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AUTH.NAME AUTHOR AFFILIATION
BOUCHEY,G.D. Washington Public Power Supply System
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards Amend 21 to FSAR.Affidavit will be furnished within 10 days of filing to reflect distribution of amend,per 10CFR2.101.

Revised 3/2/82 GP

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Washington Public Power Supply System
P.O. Box 968 3000 George Washington Way Richland, Washington 99352 (509) 372-5000

February 5, 1982
G02-82-146
SS-L-02-CDT-82-029

Docket No. 50-397

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
FSAR AMENDMENT NO. 21

The Washington Public Power Supply System herewith submits sixty (60) copies of Amendment 21 to its Final Safety Analysis Report.

Pursuant to 10CFR2.101, we will, within ten (10) days of filing, furnish to you an affidavit reflecting our distribution of this amendment to your designated distribution list.

Very truly yours,

G D Bouchey

G. D. Bouchey
Deputy Director, Safety and Security

CDT/jca

cc: R Auluck - NRC
WS Chin - BPA
R Feil - NRC Site

B001
s 3/60

STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

Subject: FSAR Amend. 21

I G. D. BOUCHEY, being duly sworn, subscribe to and say that I am the Deputy Director, Safety and Security, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that I have full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information and belief the statements made in it are true.

DATED 2/4, 1982

G D Bouchey
G. D. BOUCHEY, Deputy Director
Safety & Security

On this day personally appeared before me G. D. BOUCHEY to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes therein mentioned.

GIVEN under my hand and seal this 4th day of February 1982.

Rebecca C Hobson
Notary Public in and for the
State of Washington

Residing at 8304 W. Deschutes
Kennewick
1-1-86

LS

[illegible]

INSTRUCTIONS FOR INSERTING AMENDMENT NO. 21

The following instructional information and checklist is furnished to help you insert Amendment No. 21 into the Washington Public Power Supply System Project No. 2 FSAR.

Discard the old sheets and insert the new sheets as listed below (front page/back page). Keep these instructions and list of effective pages in front of Volume 1 to serve as a record of changes.

DISCARD OLD PAGE	INSERT. NEW PAGE
VOLUME 1	
1-iii/1-iv	1-iii/1-iv
1.2-29/1.2-30	1.2-29/1.2-29a; -/1.2-30
1.2-33/1.2-34	1.2-33/1.2-33a; -/1.2-34
1.2-43/1.2-44	1.2-43/1.2-43a; -/1.2-44
1.3-15/1.3-16	1.3-15/1.3-16
1.3-33/1.3-34	1.3-33/1.3-34
2-xiii/2-xiv	2-xiii/2-xiv
Figure 2.1-6	Figure 2.1-6
VOLUME 6	
2.5H-iii/-	2.5H-iii/-
Figure 2.5H-8b	Figure 2.5H-8b
	Figure 2.5H-8c
Figure 2.5H-10b	Figure 2.5H-10b
	Figure 2.5H-10c
Figure 2.5H-12b	Figure 2.5H-12b
	Figure 2.5H-12c
Figure 2.5H-14b	Figure 2.5H-14b
	Figure 2.5H-14c
Figure 2.5H-16b	Figure 2.5H-16b
Figure 2.5H-16c	Figure 2.5H-16c
Figure 2.5H-16d	Figure 2.5H-16d
	Figure 2.5H-17
	Figure 2.5H-18

DISCARD OLD PAGE	INSERT NEW PAGE
Volume 7	
3-xxxvii/3-xxxviii	3-xxxvii/3-xxxviii
3.1-77/3.1-78	3.1-77/3.1-78; 3.1-78a/-
3.2-15/3.2-16	3.2-15/3.2-16
3.2-17/3.2-18	3.2-17/3.2-18
3.2-19/3.2-20	3.2-19/3.2-20
3.2-29/3.2-30	3.2-29/3.2-30
3.2-31/3.2-32	3.2-31/3.2-32
3.4-3/3.4-4	3.4-3/3.4-4
Volume 9	
3.7-47/3.7-48	3.7-47/3.7-48
3.8-45/3.8-46	3.8-45/3.8-46
3.8-95/3.8-96	3.8-95/3.8-96
3.8-113/3.8-114	3.8-113/3.8-114
3.8-191/3.8-192	3.8-191/3.8-192
Volume 10	
3.9-35/3.9-36	3.9-35/3.9-36; 3.9-36a/-
3.9-55/3.9-56	3.9-55/3.9-56
Volume 11	
4-vii/4-viii	4-vii/4-viii
4-ix/4-x	4-ix/4-x
4.6-9/4.6-10	4.6-9/4.6-10
4.6-11/4.6-12	4.6-11/4.6-12
4.6-31/4.6-32	4.6-31/4.6-32
Figure 4.6-6b	Figure 4.6-6b
Figure 4.6-6c	Figure 4.6-6c
5.2-9/5.2-10	5.2-9/5.2-10; 5.2-10a/-
5.2-39/5.2-40	5.2-39/5.2-39a; -/5.2-40
5.2-41/5.2-42	5.2-41/5.2-42
5.2-45/5.2-46	5.2-45/5.2-46
Figure 5.2-4	Figure 5.2-4
Figure 5.2-5	Figure 5.2-5
Figure 5.2-6	Figure 5.2-6
5.4-23/5.4-24	5.4-23/5.4-24
5.4-25/5.4-26	5.4-25/5.4-26

DISCARD OLD PAGE	INSERT NEW PAGE
5.4-27/5.4-28	5.4-27/5.4-28
5.4-35/5.4-36	5.4-35/5.4-36
VOLUME 12	
6-v/6-va	6-v/6-va
6.2-29/6.2-30	6.2-29/6.2-30
6.2-30a/-	6.2-30a/-
6.2-33/6.2-33a	6.2-33/6.2-33a
6.2-57/6.2-58	6.2-57/6.2-58; 6.2-58a/-
6.2-75/6.2-76	6.2-75/6.2-76
6.2-87/6.2-88	6.2-87/6.2-88
6.2-117/6.2-118	6.2-117/6.2-118
Figure 6.2-17b	Figure 6.2-17b
6.4-7/6.4-7a	6.4-7/6.4-7a
VOLUME 13	
7-iii/7-iv	7-iii/7-iv
7-v/7-vi	7-v/7-vi
7-ix/7-x	7-ix/7-x
7.1-9/7.1-10	7.1-9/7.1-10
7.2-9/7.2-10	7.2-9/7.2-10; 7.2-10a/-
7.2-13/7.2-14	7.2-13/7.2-14
7.3-5/7.3-6	7.3-5/7.3-6
7.3-21/7.3-22	7.3-21/7.3-22
7.3-23/7.3-24	7.3-23/7.3-23a; -/7.3-24
7.3-79/-	7.3-79/-
VOLUME 14	
7.5-7/7.5-8	7.5-7/7.5-8
7.5-9/7.5-10	7.5-9/7.5-10
7.6-1/7.6-2	7.6-1/7.6-2
7.6-9/7.6-10	7.6-9/7.6-9a; -/7.6-10
7.6-13/7.6-14	7.6-13/7.6-14
7.6-15/7.6-16	7.6-15/7.6-15a; -/7.6-16
7.6-39/7.6-40	7.6-39/7.6-40
7.6-41/7.6-42	7.6-41/7.6-42
	7.6-49/-
Figure 7.6-3	Figure 7.6-3
7.7-1/7.7-2	7.7-1/7.7-2

DISCARD OLD PAGE	INSERT NEW PAGE
7.7-43/7.7-44	7.7-43/7.7-44
7.7-45/7.7-46	7.7-45/7.7-45a; -/7.7-46; 7.7-46a/-
7.7-47/7.7-48	7.7-47/7.7-48
7.7-49/7.7-50	7.7-49/7.7-50
VOLUME 15	
8.3-77/8.3-78	8.3-77/8.3-78
VOLUME 16	
9-i/9-ii	9-i/9-ii
9.1-23/9.1-24	9.1-23/9.1-23a; -/9.1-24
9.1-25/9.1-26	9.1-25/9.1-26; 9.1-26a/9.1-26b; 9.1-26c/-
9.1-27/9.1-28	9.1-27/9.1-28
9.1-65/9.1-66	9.1-65/9.1-66
9.2-27/9.2-27a	9.2-27/9.2-27a
9.2-29/9.2-30	9.2-29/9.2-30
9.2-31/9.2-32	9.2-31/9.2-31a; -/9.2-32
9.2-33/9.2-34	9.2-33/9.2-34
9.2-39/9.2-40	9.2-39/9.2-40; 9.2-40a/-
9.2-41/9.2-42	9.2-41/9.2-41a; -/9.2-42
9.3-3/9.3-4	9.3-3/9.3-4
9.3-33/9.3-34	9.3-33/9.3-34
9.4-53/9.4-54	9.4-53/9.4-53a; -/9.4-54
9.4-55/9.4-56	9.4-55/9.4-56
9.4-79/9.4-80	9.4-79/9.4-80
9.5-43/9.5-44	9.5-43/9.5-43a; -/9.5-44
VOLUME 17	
10-iii/10-iv	10-iii/10-iv
10.4-17/10.4-17a	10.4-17/10.4-17a
10.4-17b/10.4-17c	10.4-17b/10.4-17c
-/10.4-18	10.4-17d/10.4-18
11.5-13/11.5-14	11.5-13/11.5-14

DISCARD OLD PAGE	INSERT NEW PAGE
VOLUME 18	
15.0-22/- (Figure 15.0-2)	15.0-22/- (Figure 15.0-2)
15.1-1/15.1-2	15.1-1/15.1-2
15.1-27/15.1-28	15.1-27/15.1-28
VOLUME 19	
15.2-19/15.2-20	15.2-19/15.2-20
15.2-39/15.2-40	15.2-39/15.2-40
15.2-65/15.2-66	15.2-65/15.2-66
15.2-67/15.2-68	15.2-67/15.2-68
15.2-71/15.2-71a	15.2-71/15.2-71a
17.1-1/17.1-2	17.1-1/17.1-2
17.1-2a/-	17.1-2a/-
VOLUME 20	
C.2-43/C.2-44	C.2-43/C.2-44
C.2-45/C.2-46	C.2-45/C.2-46
C.3-41/C.3-42	C.3-41/C.3-41a; -/C.3-42
C.3-57/C.3-58	C.3-57/C.3-58
H.1.2-3/H.1.2-4	H.1.2-3/H.1.2-4; H.1.2-4a/-
VOLUME 21	
LIST OF NRC QUESTIONS	QUESTION/SECTION CROSS REFERENCE
010.9-1/010.10-1	010.9-1/010.010-1
010.021-1/010.22-1	010.021-1/010.022-1
010.27-1/010.28-1	010.27-1/010.028-1
010.33-1/010.34-1	010.33-1/010.034-1
	010.035-1/010.036-1
	010.037-1/010.038-1
	010.039-1/010.040-1; 010.040-2/-
	010.041-1/010.041-2;
	010.041-3/010.041-4;
	010.041-5/010.041-6;
	010.041-7/010.041-8
	-/010.042-1

DISCARD OLD PAGE	INSERT NEW PAGE
	010.042-2/-
	-/010.044-1
	010.045-1/010.046-1
	010.047-1/010.048-1
	010.049-1/010.049-2;
	010.049-3/010.049-4;
	010.049-5/-
	Figure 010.049-1
	Figure 010.049-2
	-/010.050-1
	010.051-1/010.052-1
	010.053-1/-
	-/010.056-1;
	010.056-2/-
	010.057-1/010.058-1
	010.059-1/010.060-1
	010.061-1/010.062-1
	010.063-1/010.064-1
	010.065-1/-
Figure 022.062-1	Figure 022.062-1
022.063-3/-	022.063-3/022.064-1
	022.065-1/022.066-1
	022.067-1/022.068-1
	022.069-1/022.070-1
	022.071-1/022.072-1
	022.073-1/022.074-1
	022.075-1/022.076-1
	022.077-1/022.078-1
	022.079-1/022.080-1
	022.081-1/022.082-1
	022.083-1/022.084-1
	022.085-1/022.086-1
	022.087-1/022.088-1
	022.089-1/022.090-1
	022.091-1/022.092-1
	022.093-1/022.093-2;
	022.093-3/022.094-1;
	022.094-2/-
	022.095-1/022.096-1
	022.097-1/022.098-1
	022.099-1/022.099-2

DISCARD OLD PAGE	INSERT NEW PAGE
	-/022.100-1; 022.100-2/- 022.101-1/022.101-2 -/022.102-1 022.103-1/022.104-1 022.105-1/022.106-1
031.070-2/031.070-3 031.070-4/031.070-5 031.113-1/-	031.070-2/031.070-3 031.070-4/031.070-5 031.113-1/031.114-1; 031.114-2/- 031.115-1/031.116-1 031.119-1/031.120-1 031.121-1/031.122-1 031.123-1/031.124-1; 031.124-2/- 031.125-1/031.126-1; 031.126-2/- 031.127-1/- 031.129-1/031.130-1 031.131-1/031.131-2 -/031.132-1 031.133-1/031.134-1
Remove 40 Series Questions and place at beginning of Volume 22	
<u>VOLUME 22</u>	
	Insert 40 Series Questions from Volume 21
040.055-1/040.056-1	040.055-1/040.056-1 040.081-1/040.081-2 -/040.082-1; 040.082-2/040.082-3 040.085-1/040.086-1; 040.086-2/- 040.087-1/- 040.089-1/-

DISCARD OLD PAGE	INSERT NEW PAGE
110.033-1/110.034-1	110.033-1/110.034-1
211.005-1/211.006-1; 211.006-2/211.006-3; 211.006-4/211.006-5; Figure 211.006-1 Figure 211.006-2	211.005-1/211.006-1
211.009-1/211.010-1	211.009-1/211.010-1
211.010-2/-	211.010-2/-
-/211.018-1	-/211.018-1
-/211.032-1	-/211.032-1
211.087-1/211.087-2	211.087-1/211.087-2; 211.087-3/211.087-4; 211.087-5/- Figure 211.087-1
Remove 211.101-1 through 211.210-1 and place at beginning of Volume 23	
VOLUME 23	
	Insert 211.101-1 thru 211.210-1
	211.107-1/211.108-1
	211.109-1/211.110-1
	211.111-1/211.112-1
211.113-1/-	211.113-1/211.113-2
	-/211.114-1
211.115-1/-	211.115-1/211.116-1; 211.116-2/211.116-3
	211.117-1/211.118-1
	211.119-1/211.120-1; 211.120-2/-
	211.121-1/211.122-1
-/211.124-1	211.123-1/211.124-1
	211.125-1/211.126-1
	-/211.128-1; 211.128-2/-
	-/211.130-1

DISCARD OLD PAGE	INSERT NEW PAGE
	211.131-1/211.132-1;
	211.132-2/-
	211.133-1/211.134-1
	211.135-1/-
211.137-1/-	211.137-1/211.138-1
	211.143-1/211.143-2
	-/211.144-1;
	211.144-2/-
211.149-1/-	211.149-1/211.150-1
211.151-1/-	211.151-1/211.152-1
211.163-1/-	211.163-1/211.164-1;
	211.164-2/-

DISCARD OLD PAGE	INSERT NEW PAGE
211.167-1/-	211.167-1/211.168-1
	211.171-1/211.171-2;
	211.171-3/211.171-4
	211.173-1/211.174-1
211.179-1/-	211.179-1/211.180-1
211.181-1/-	211.181-1/211.182-1
-/211.186-1	211.185-1/211.186-1
	211.189-1/211.189-2
	-/211.196-1
	-/211.198-1;
	211.198-2/-
	211.199-1/211.199-2
	211.201-1/211.202-1;
	211.202-2/-
	211.203-1/211.204-1
	-/211.206-1;
	211.206-2/211.206-3;
	211.206-4/211.206-5;
	211.206-6/211.206-7;
	211.206-8/211.206-9;
	211.206-10/211.206-11
211.207-1/211.207-2	211.207-1/211.207-2
	-/211.208-1
	211.211-1/211.212-1
	211.213-1/-
	Figure 211.213-1
VOLUME 23	
	(Tab for 271 Series will be supplied in Amendment No. 23)
	271.001-1/271.001-2;
	271.001-3/271.002-1;
	271.002-2 through 271.002-269
	271.003-1/271.003-2;
	271.003-3/-
	-/271.006-1

DISCARD OLD PAGE	INSERT NEW PAGE
	Tab for 281 Series 281.001-1/281.001-2 -/281.002-1 281.003-1/281.004-1 281.005-1/281.006-1 281.007-1/281.008-1 281.009-1/281.009-2 -/281.010-1 281.011-1/281.012-1 281.013-1/281.014-1
321.003-1/321.004-1	321.003-1/321.004-1
	-/360.006-1 360.006-2/- 360.007-1/360.007-2; 360.007-3/360.008-1; 360.008-2/360.008-3; 360.008-4/- 360.009-1/360.009-2; 360.009-3/360.009-4; 360.009-5/360.009-6; 360.009-7/360.010-1; 360.010-2/360.010-3 360.011-1/360.011-2 -/360.012-1; 360.012-2/- 360.013-1/360.013-2; 360.013-3/-
	Tab for 361 Series 361.001-1/361.002-1; 361.002-2/- 361.003-1/361.004-1; 361.004-2/361.004-3; 361.004-4/361.004-5; 361.004-6/361.004-7; 361.004-8/361.004-9; 361.004-10/361.004-11; 361.004-12/361.004-13; 361.004-14/361.004-15; 361.004-16/361.004-17;

DISCARD OLD PAGE	INSERT NEW PAGE
	361.004-18/361.004-19;
	361.004-20/361.004-21;
	361.004-22/361.004-23;
	361.004-24/361.004-25;
	361.004-26/361.004-27;
	361.004-28/-
	361.005-1/361.005-2;
	361.005-3/361.005-4;
	361.005-5/361.005-6;
	-/361.006-1
	Figure 361.006-1
	361.007-1/361.007-2
	-/361.008-1;
	361.008-2/-
	361.009-1/361.009-2;
	361.009-3/361.010-1;
	361.010-2/361.010-3
	Figure 361.010-1
	361.011-1/361.011-2
	-/361.012-1
	361.013-1/361.013-2
	-/361.014-1
	-/362.010-1;
	362.010-2/362.010-3;
	362.010-4/362.010-5
	-/362.012-1;
	362.012-2/-
	362.013-1/362.014-1;
	362.014-2/362.014-3
	Figure 362.014-1
Remove 421 Series through 441 Series Questions and place at beginning of Volume 24 (provided)	
VOLUME 24 (new)	
	Insert 421 thru 441 Series Questions
421.007-1/421.008-1	421.007-1/421.008-1
421.041-1/421.042-1	421.041-1/421.041-2
	-/421.042-1
	432.017-1/432.017-2
	-/432.018-1
	432.019-1/432.020-1
	432.021-1/432.022-1
	432.023-1/432.024-1
	432.025-1/-

DISCARD OLD PAGE	INSERT NEW PAGE
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	Letter to be added at end of ~~Appendix 4~~
FIRE PROTECTION EVALUATION	
	Add to end of F.3: Exhibit A.1.1 (M519) Exhibit A.1.2 (M521) Exhibit A.1.3 (M524) Exhibit A.1.5 (M548)

DISCARD OLD PAGE	INSERT NEW PAGE
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	Exhibit A.1.7 (M551)
	Exhibit A.1.8 (M529)
	Exhibit A.1.9 (M543)
	Exhibit A.1.10 (M521)
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c. Limit the release of radioactive materials by closing the primary containment barrier in case of a major leak from the nuclear system inside the primary containment.

1.2.2.5.12 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the reactor vessel before the main steam line isolation valves are closed in case of a main steam line break outside the primary containment.

1.2.2.5.13 Main Steam Line Radiation Monitoring System

The main steam line radiation monitoring system consists of four gamma radiation monitors located externally to the main steam lines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used to initiate a reactor scram and to close the main steam line isolation valves.

1.2.2.5.14 Standby Service Water and HPCS Service Water Systems

The standby service water system consists of two 100 percent redundant systems. Each system consists of a spray pond and pump and piping supplying the associated residual heat removal system heat exchanger, standby diesel generator, essential HVAC coolers, RHR and LPCS pump coolers, sample coolers, and post-LOCA hydrogen recombiners.

Cooling water is supplied during a postulated loss of coolant accident to the RHR heat exchangers to remove heat when the containment cooling mode of the RHR system is placed in operation. During normal operation, standby service water is also supplied to the RHR heat exchangers for the shutdown function of the RHR system.

The HPCS service water system shares spray pond A with the standby service water system. The pump supplies cooling water to the HPCS diesel generator and the essential HVAC coolers for the HPCS diesel generator and HPCS pump areas.

Cooling water is supplied to all diesel generator cooling systems whenever the diesel generators are started.

1.2.2.5.15 Reactor Building - Secondary Containment

The reactor building completely surrounds the primary containment. The building provides secondary containment when the primary containment is closed and in service, and serves as a primary containment during periods when the primary containment is open, such as during refueling. The reactor building also houses refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor safety and auxiliary systems.

The design of the reactor building includes provisions for seismic load resistance and low infiltration and exfiltration rates. The building consists of poured-in-place, reinforced-concrete exterior walls up to the refueling floor. Above this level, the building structure is steel frame with insulated metal siding with sealed joints. Access to the building is through interlocked double doors.

1.2.2.5.16 Reactor Building Ventilation Exhaust Radiation Monitoring System

The reactor building ventilation exhaust radiation monitoring system consists of a number of radiation monitors arranged to monitor the activity level of the ventilation exhaust from the reactor building and primary containment. Upon detection of high radiation, the reactor building is automatically isolated and the standby gas treatment system is started.

1.2.2.5.17 Standby Gas Treatment System

The standby gas treatment system consists of two identical filter trains. Each filter train consists of a filter unit, two fans, ductwork and associated valves.

Either filter train may be considered as an installed spare with the other train capable of passing the required amount of air. Either train alone is capable of exchanging the total reactor building volume once in a 24-hour period.

Each filter unit contains electric heaters, a prefilter, high-efficiency particulate filters (water and fire resistant), an iodine filter (high ignition temperature), and instrumentation to measure temperature and flow.

The system maintains a slightly negative internal building pressure and can process all gaseous effluent prior to its discharge from the reactor building.

LOCA. This is accomplished by directing the leakage through the closed main steam line isolation valves to a bleed line into an area served by the SGTS. The flow is effected by a blower which directs the leakage into the reactor building and eventually through the standby gas treatment system. Thus, leakages through the MSIV will be processed by the SGTS prior to release to the atmosphere.

1.2.2.6 Power Conversion System

1.2.2.6.1 Turbine-Generator

The turbine is an 1800-rpm, tandem-compound (one double-flow high pressure turbine and three double flow low pressure turbines), reheat unit with an electrohydraulic governor for normal operation. The turbine-generator is provided with an emergency trip system for turbine overspeed. The rating of the turbine-generator is 1,154,745 kW at 2.5 in. Hg abs.

The generator is a direct-driven, three-phase, 60-Hertz, 25,000-volt, 1800-rpm, hydrogen inner-cooled, synchronous generator rated at 1,230 MVA at 0.975 power factor, 0.58 short circuit ratio at a maximum hydrogen pressure of 75 psig.

1.2.2.6.2 Main Steam System

The main steam system consists of four 26-inch diameter lines (which expand to 30-inch diameter lines inside the turbine building) extending from the outermost main steam line isolation valves to the main turbine stop valves. The use of four main steam lines permits testing of the turbine stop valves and main steam line isolation valves during station operation with only a minimum of load reduction. The design pressure and temperature of the main steam system from the outermost MSIV to the turbine stop valve is 1250 psig at 575^{o}F. Other features include drains and parts of the turbine bypass system.

1.2.2.6.3 Main Condenser

The main condenser is a triple-pressure, single-pass, deaerating-type condenser with a divided water box. The condenser includes provisions for accepting up to 25% of the main steam flow at design conditions from the turbine bypass system and serves as a heat sink for several other flows, such as exhaust steam from the feed pump turbines, cascading heater drains, feedwater heater shell operating vents, and condensate pump suction vents.

1.2.2.6.4 Main Condenser Evacuation System

The main condenser evacuation system is designed to remove noncondensable gases from the condenser, including air inleakage and dissociation products originating in the reactor, and to continuously exhaust them to the gaseous radwaste system during operation. The system consists of two 100%-capacity, twin-element first stage and single-element second stage steam-jet air ejector (SJAE) units complete with intercondensers for normal plant operation and a mechanical vacuum pump for use during startup. Discharge from the vacuum pumps during startup is routed to the elevated release point.

1.2.2.6.5 Turbine Gland Seal System

The turbine gland seal system is designed to provide a means of preventing air leakage into or radioactive steam leakage out of the turbine. The system consists of two 100% steam evaporators, steam seal pressure regulators, steam seal header, gland seal steam condenser and blowers, and the associated piping, valves, and instrumentation.

1.2.2.6.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The capacity of the turbine bypass system is 25% of the turbine design steam flow. The pressure regulation system provides main turbine control valve and bypass valve position demands so as to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power level by changing reactor recirculation flow rate.

1.2.2.6.7 Circulating Water System

The circulating water system provides the condenser with a continuous supply of cooling water. It is a closed system utilizing forced draft cooling towers. Makeup water to the system is provided from makeup pumps located in an intake structure on the Columbia River. The makeup water replaces the water lost by evaporation, drift, and blowdown.

1.2.2.9.6 Gaseous Radwaste System Control

Gaseous radwastes are discharged through a reactor building elevated release point. Radiation levels of the release are continuously monitored and recorded. Isolation of the main condenser off-gas is automatically initiated prior to release should the activity of the off-gas exceed discharge limits.

1.2.2.10 Shielding

The shielding in the plant is designed to minimize exposure of plant personnel to radiation. The radiation levels during operation or shutdown conditions have been considered in determining the shielding requirements.

1.2.2.11 Fuel Handling and Storage Systems

1.2.2.11.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient coolant and shielding are maintained to prevent overheating and excessive personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherance to Seismic Category I requirements, and prevention of k_{eff} from reaching 0.90 under dry conditions or 0.95 under flooded conditions.

1.2.2.11.2 Fuel Handling System

The fuel handling equipment includes a fuel inspection stand, fuel preparation machine, a 125-ton crane, a refueling platform, a new fuel transfer basket, jib cranes, and other related tools for fuel and reactor servicing.

1.2.2.11.3 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup subsystem provides the removal of decay heat from stored spent fuel and maintains specified water temperature, purity, clarity, and level. This prevents spent fuel overheat and the buildup of excessive radioactive materials in the cooling water, thereby minimizing possible exposures to plant personnel.

1.2.2.12 Cooling Water and Auxiliary Systems

1.2.2.12.1 Reactor Building Closed Cooling Water System

The reactor building closed cooling water system consists of pumps, heat exchangers, controls, and instrumentation to provide adequate cooling for the reactor auxiliary systems. The system is designed to provide a closed cooling water loop between non-essential systems which are potentially radioactive and the service water system.

1.2.2.12.2 Plant Service Water System

Normal plant service water is supplied from service water pumps located in the circulating water pumphouse. Two (2) service water pumps are provided. The plant service water system is designed to remove heat from various auxiliary equipment located within the plant.

1.2.2.12.3 Ultimate Heat Sink

Two spray ponds that serve as the ultimate heat sink conservatively have a combined equivalent storage of thirty days, assuming no makeup and maximum evaporation and drift losses. Provisions are made to replenish the sink to allow continued cooling capability beyond the initial thirty day period.

1.2.2.12.4 Demineralized Water Makeup System

The demineralized water makeup system is comprised of the plant makeup water treatment system and the demineralized water system. The plant makeup treatment system is designed to provide treated water for potable and sanitary systems and demineralized water to the demineralized water system.

The demineralized water system is designed to provide demineralized water to the condensate storage tanks for plant makeup and demineralized water for other plant operating requirements.

1.2.2.12.5 Potable Water and Sanitary Drain Systems

The plant potable water system provides water for drinking and sanitary purposes. Potable water is supplied by the plant makeup water treatment system from the Columbia River.

TABLE 1.3-3 (Continued)

EMERGENCY CORE COOLING SYSTEMS (Continued)	WNP-2 BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
Low Pressure Coolant Injection[b]			
Number of loops	3	2	3
Number of pumps	3	4	3
Flow rate, gpm/pump	7450 at 26 psid	7700 at 20 psid	5050 at 20 psid
Residual Heat Removal System (See Section 5.4.7)			
Reactor Shutdown Cooling Mode:			
Number of loops	2	2	2
Number of pumps	2	4	2
Flow rate, gpm/pump[c]	7450	7700	5050
Duty, Btu/hr/heat exchanger[d]	41.6×10^6	32×10^6	30.8×10^6
Number of heat exchangers	2	2	2
Primary containment cooling mode:			
Flow rate, gpm	7450[e]	30,800	5050[e]

TABLE 1.3-3 (Continued)

EMERGENCY CORE COOLING SYSTEMS (Continued)	WNP-2 BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
Standby Service Water System (See Section 9.2.7)			
Flow rate, gpm/heat exchanger	7400	8000	5000
Number of pumps	3[f]	4	4
Reactor Core Isolation Cooling System (See Section 5.4.6)			
Flow rate, gpm	600 at 1150 psid	400 at 1120 psid	400 at 1120 psid
Fuel Pool Cooling and Cleanup System (See Section 9.1.3)			
Capacity, Btu/hr	7.6×10^6	5.7×10^6	6.6×10^6

---

[a] High-pressure coolant injection system utilized

[b] A mode of the RHR system

[c] Capacity during reactor flooding mode with more than one pump running

[d] Heat exchanger duty at 20 hours following reactor shutdown

[e] Flow per heat exchanger

[f] Includes HPCS Service Water Pumps

TABLE 1.3-8 (Continued)

ITEM	CHANGE	REASON FOR CHANGE	FSAR PORTION IN WHICH CHANGE IS DISCUSSED
Cable Chase Fire Protection System	Added dry pipe preaction system for cable chase and diesel-generator building corridor.	To protect divisional cable concentrations in these areas.	9.5.1.2.1.3
Office Area and Circulating Water Pump House Fire Protection Systems	Added wet pipe sprinkler systems for office area and circulating water pump house	To protect occupational and equipment areas	9.5.1.2.1.2
500 kV Line	Hookstick changed to motor operated switch	Available standard switches are supplied with motor operators	Fig. 8.1-9a
500 kV Line	Line terminates at H.J. Ashe Swtichyard rather than Hanford Switching Station	BPA revisions to 500 kV grid	8.1.2
230 kV Line	Deleted hookstick and 230 kV OCB at Plant Switchyard	OCB relocated to H.J. Ashe Switchyard	Fig. 8.1-9a
115 kV Line	Replaced Circuit Interrupter with 115 kV OCB at Plant Switchyard	Equipment availability	Fig. 8.1-9a
Backup Source	Utilized to supply essential loads during diesel-generator testing	PSAR did not consider partic-ulars of D.G. testing	8.3.1.1.8.1.7
Diesel-Generator Starting	Deleted automatic starting due to startup or backup transformer undervoltage	Class 1E bus undervoltage is the only undervoltage condi-tion requiring D.G. start	8.3.1.1.8.1.7 8.3.1.1.8.2.7
Diesel-Generator Trips During Emergency Operation	Added incomplete sequence trip to Division 1 and 2 DG's	Incomplete sequence indicates a D.G. malfunction having an imminent possibility of unit damage.	8.3.1.1.8.1.8

TABLE 1.3-8 (Continued)

ITEM	CHANGE	REASON FOR CHANGE	FSAR PORTION IN WHICH CHANGE IS DISCUSSED
125V, 250V DC Battery Capability	Revised supply capability from 4 hours to 2 hours	Increased dc loads	8.3.2
125V, 250V DC Charger Capability	Revised recharge capability from 8 hours to 24 hours	Increased dc loads	8.3.2
Spare 125V DC Charger	Spare charger serves as a backup for Division 1 and 2 only	Spare charger is too large to provide backup to Division 3	8.3.2
Communication Systems	The Microwave System and the commercial telephone exchange system are not redundant	Redundancy not required. (However, some plant communication functions may be served by either system)	8.2.1.5

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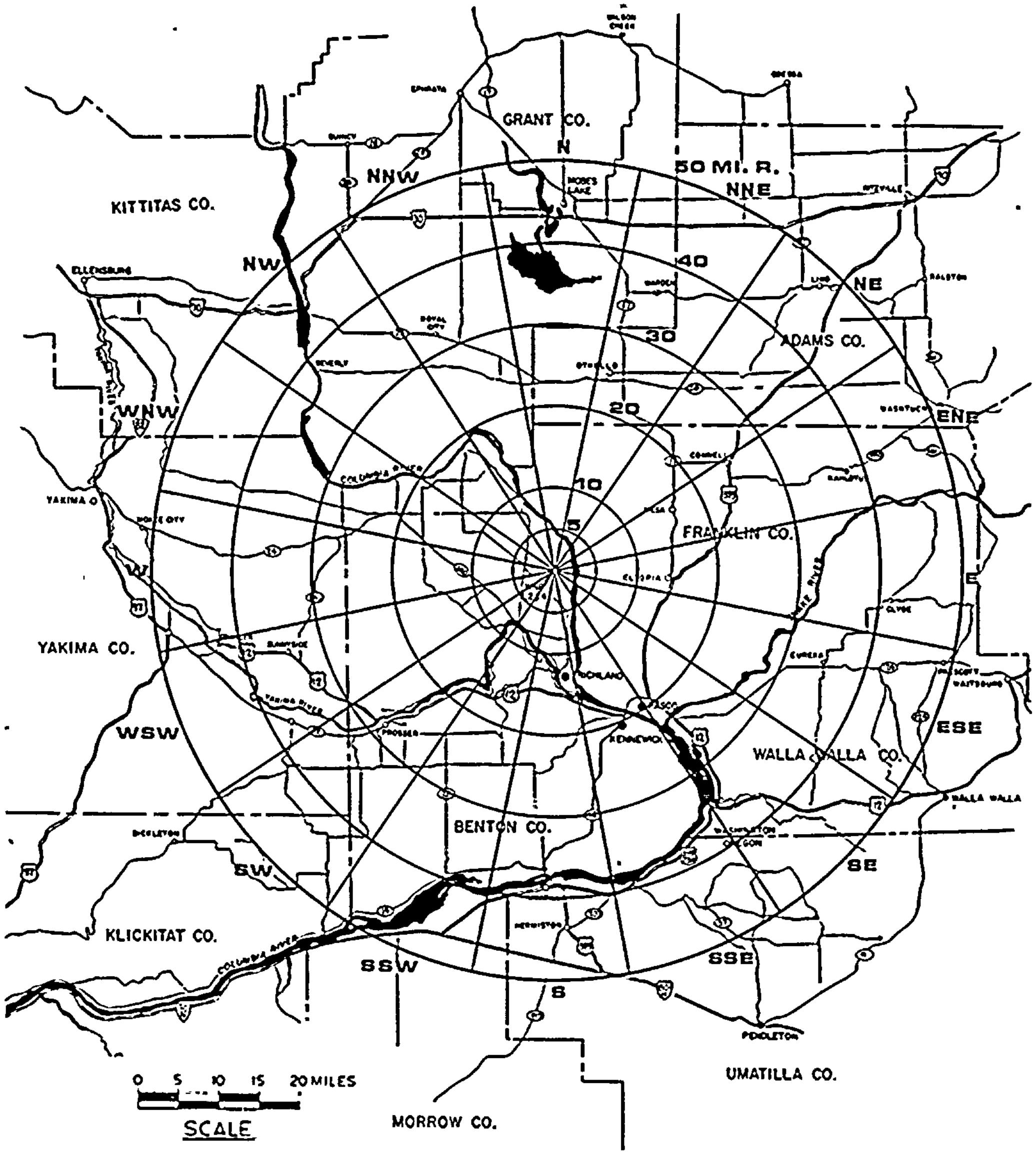

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROJECT AREA MAP - 50-MILE RADIUS	FIGURE 2.1-6

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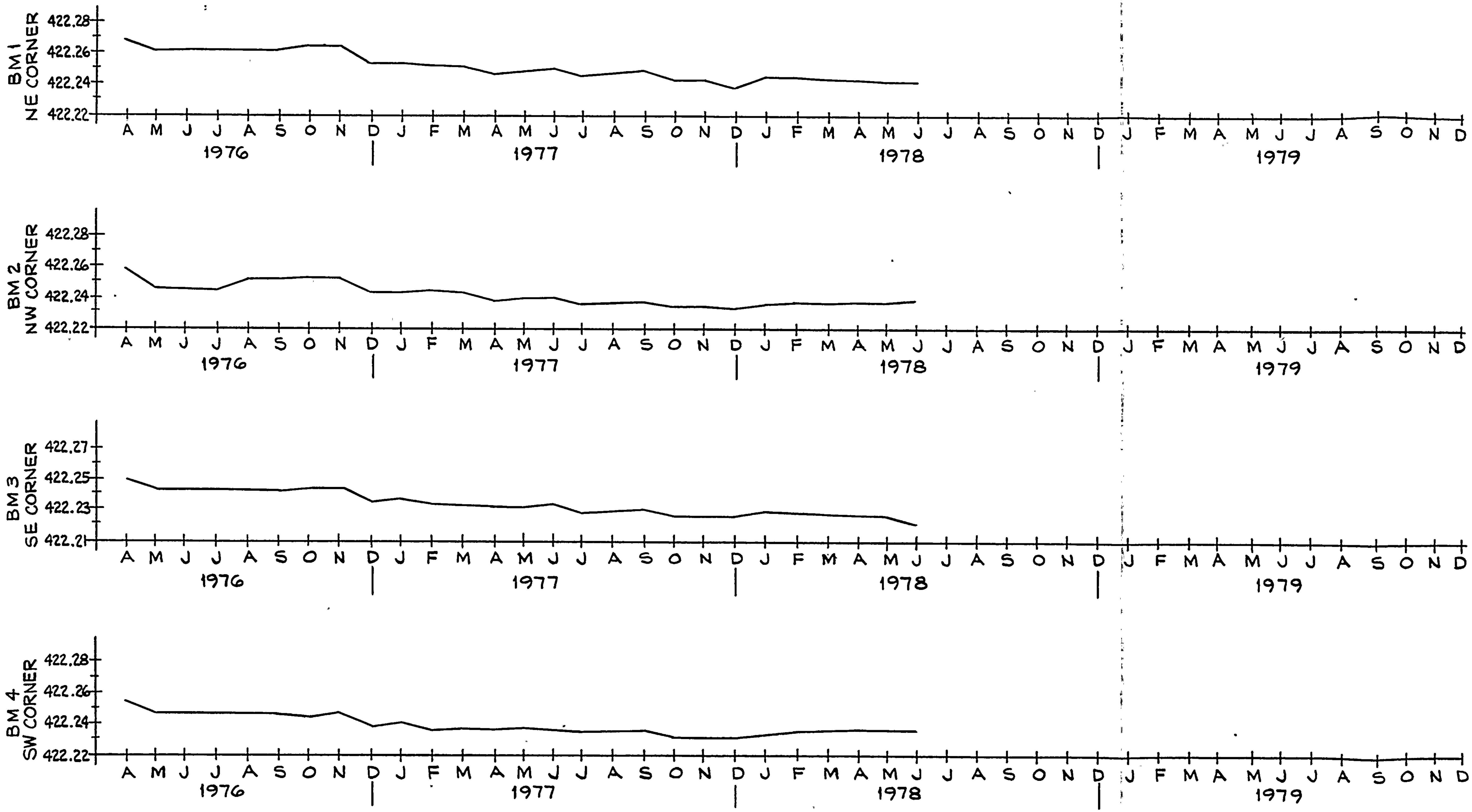
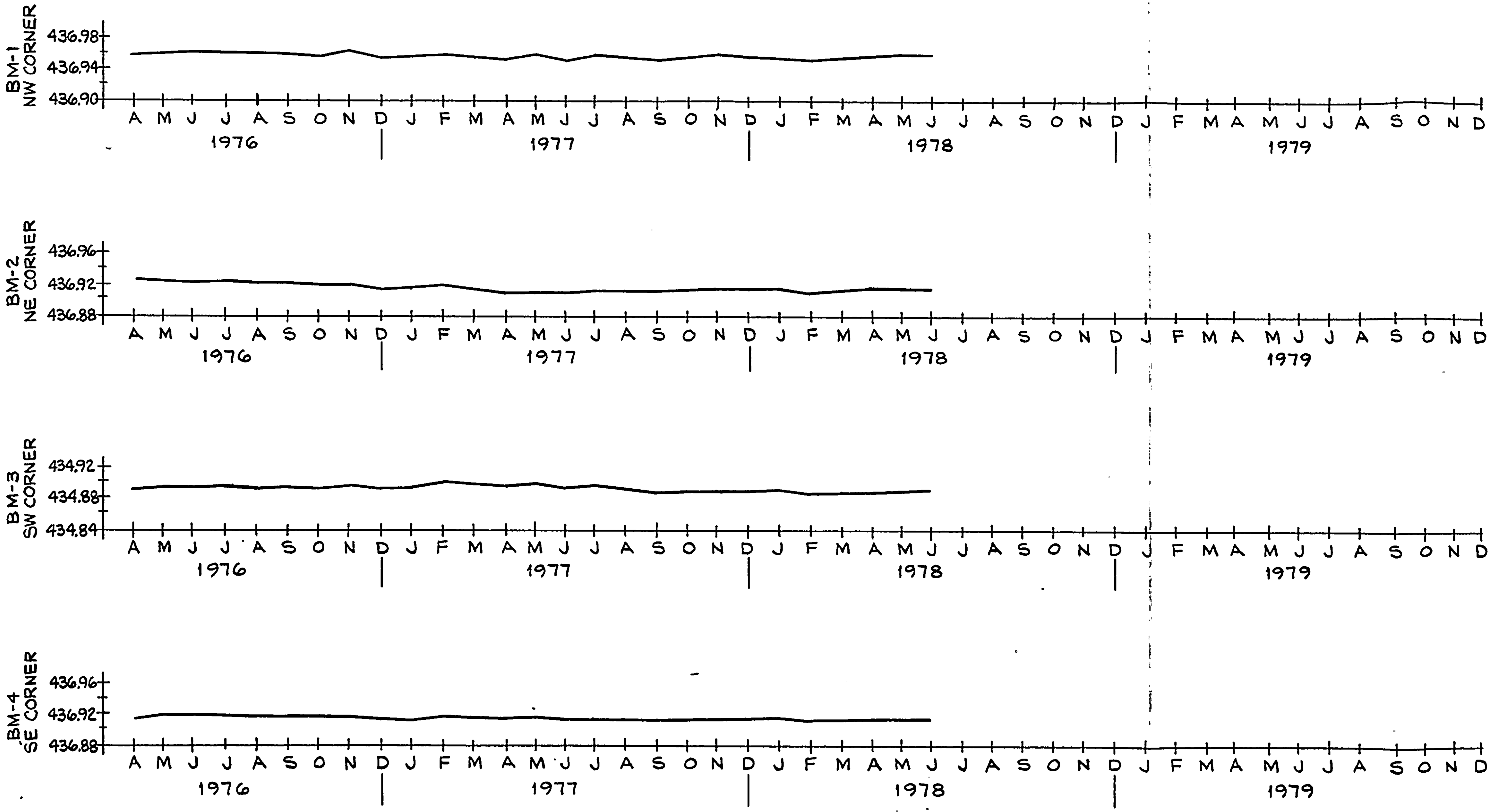
March 1979



REFER TO FIGURE 2.5H-7 FOR BENCH MARK LOCATIONS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING DATA, REACTOR BUILDING	FIGURE 2.5H- 8b



REFER TO FIGURE 2.5H-9 FOR BENCH MARK LOCATIONS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING DATA, RADWASTE AND CONTROL BUILDING	FIGURE 2.5H-10b

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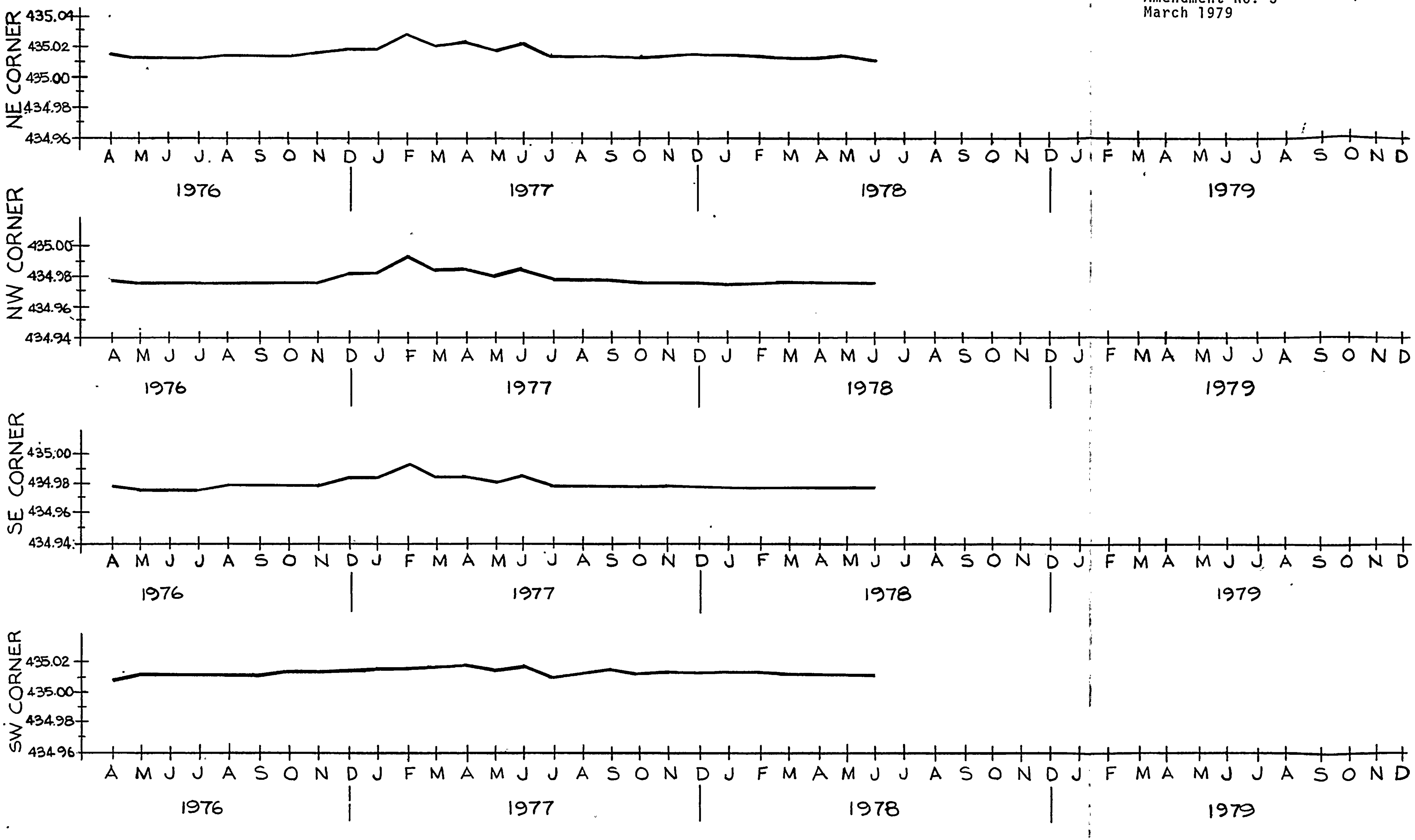
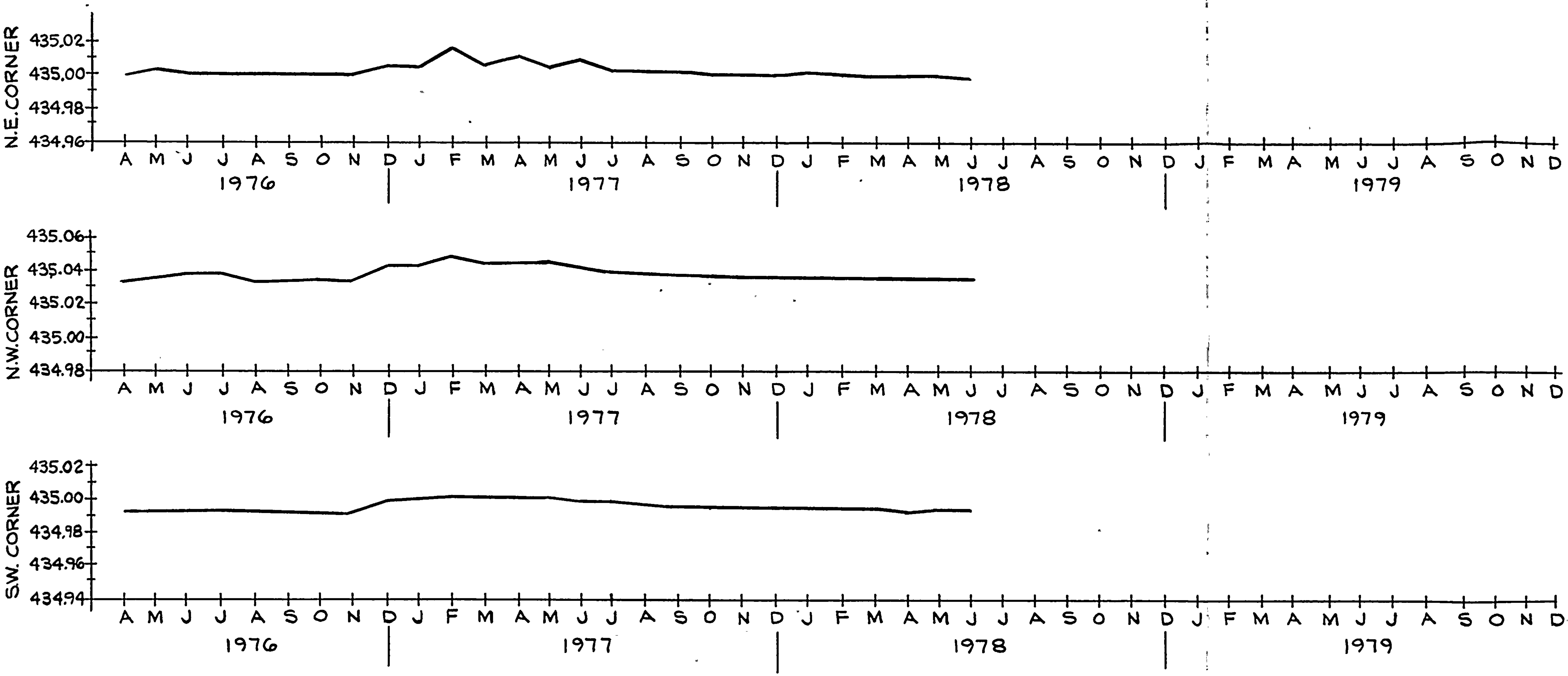
March 1979



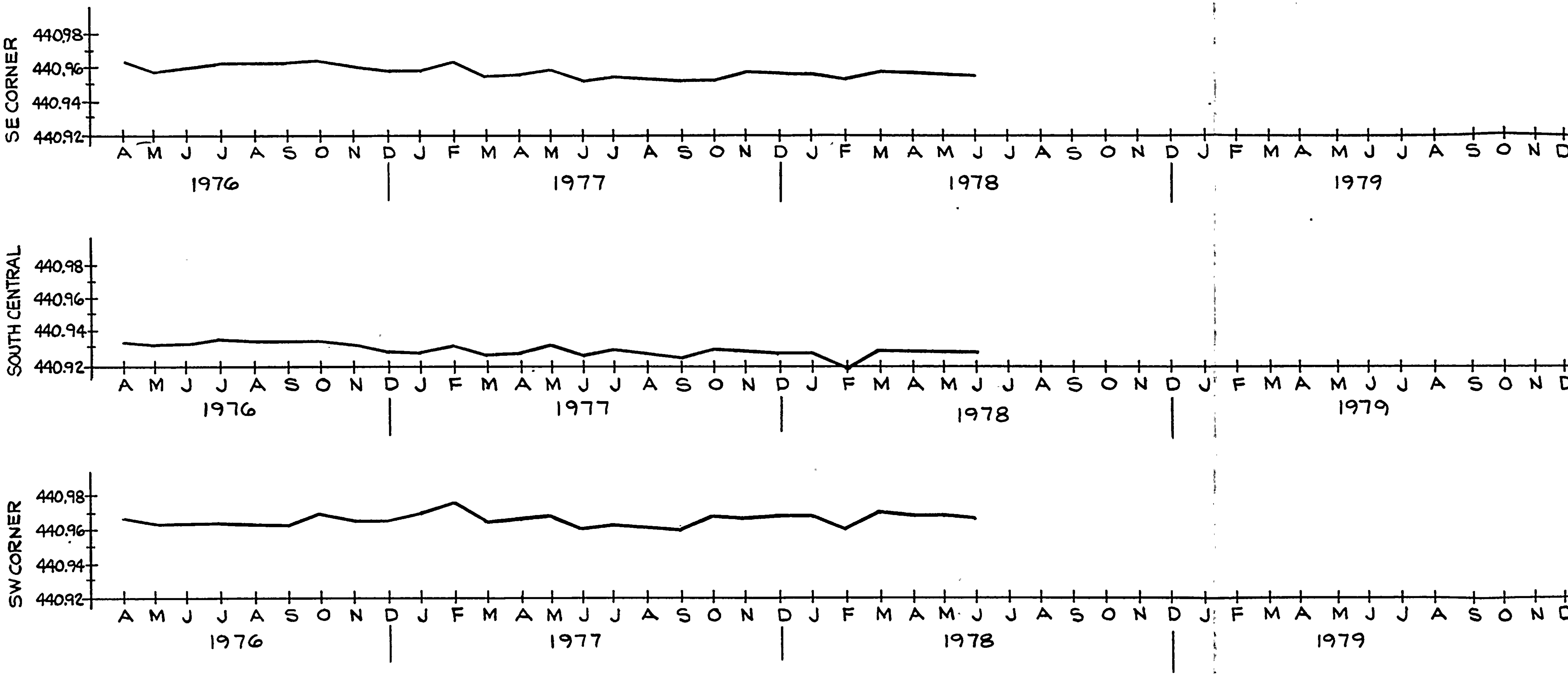
REFER TO FIGURE 2.5H-11 FOR BENCH MARK LOCATION

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING DATA, SPRAY POND 1A	FIGURE 2.5H-12b



REFER TO FIGURE 2.5H-13 FOR BENCH MARK LOCATIONS

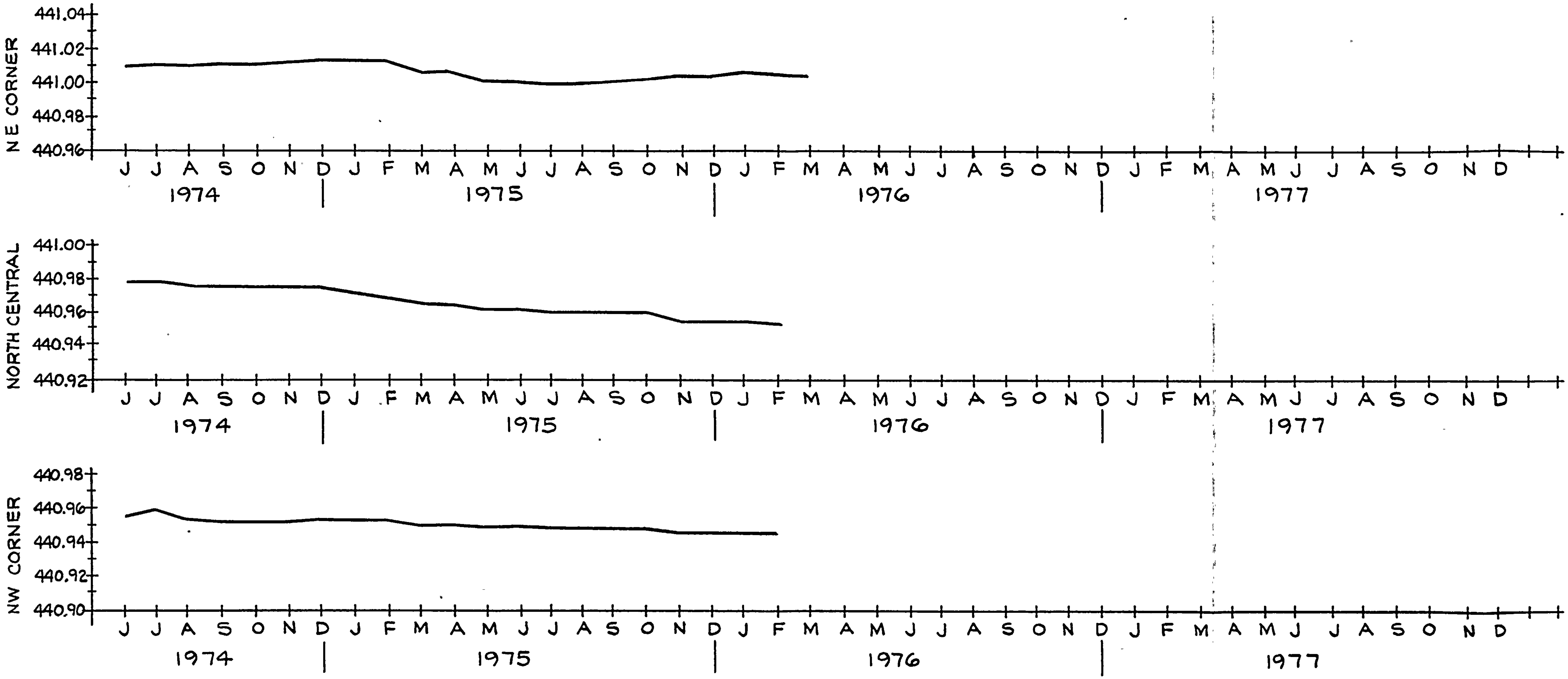
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING DATA SPRAY POND 1B	FIGURE 2.5H 14b



REFER TO FIGURE 2.5H-15 FOR BENCH MARK LOCATIONS.

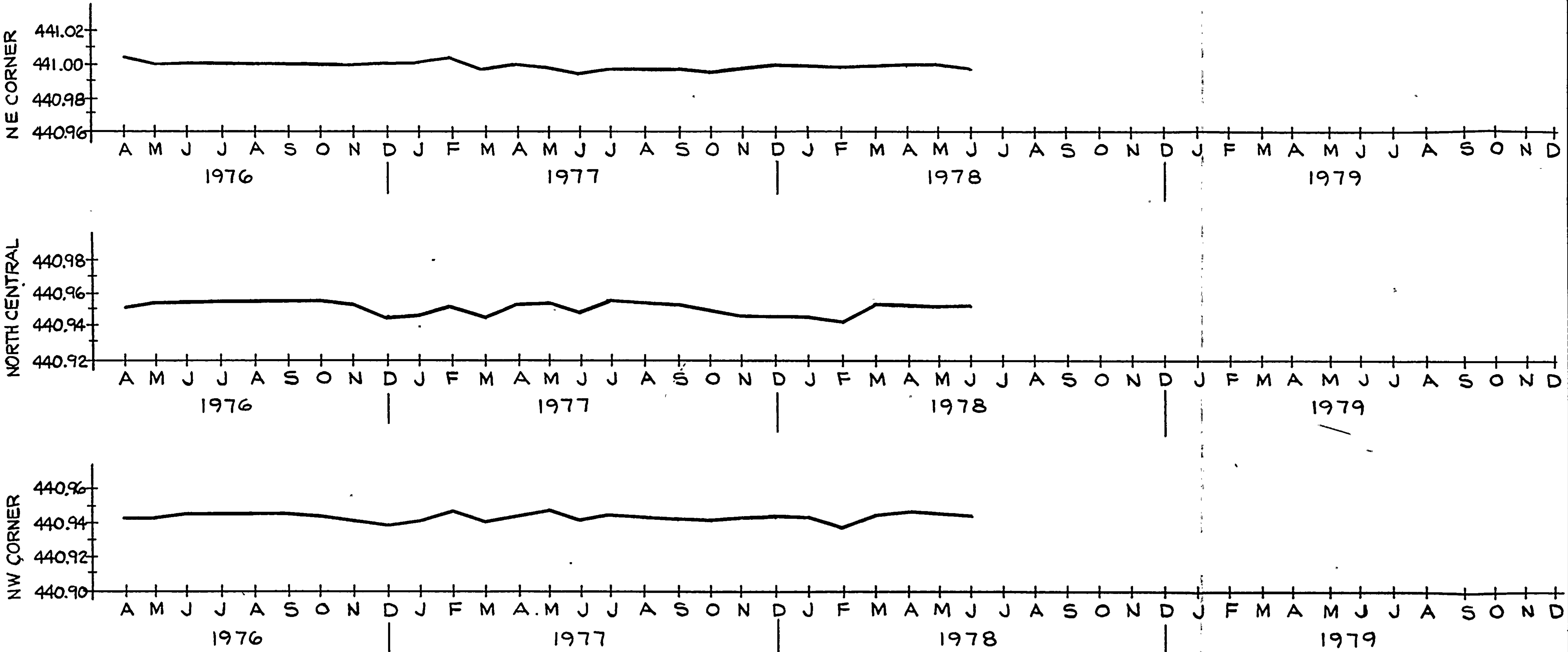
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING BENCH MARKS, TURBINE GENERATOR BUILDING	FIGURE 2.5H-16b

Amendment No. 3



REFER TO FIGURE 2.5H-15 FOR BENCH MARK LOCATIONS

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING BENCH MARKS, TURBINE GENERATOR BUILDING	FIGURE 2.5H-16c



REFER TO FIGURE 2.5H-15 FOR BENCH MARK LOCATION

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SETTLEMENT MONITORING BENCH MARKS, TURBINE GENERATOR BUILDING	FIGURE 2.5H- 16d

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i. Radiation Protection 12.0

j. Accident Analyses 15.0

3.1.2.6.2 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed, with a capacity to permit appropriate periodic inspection and testing of components important to safety, with suitable shielding for radiation protection, with appropriate containment, confinement, and filtering systems, and residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and to prevent significant reduction in fuel storage coolant inventory under accident conditions.

3.1.2.6.2.1 Evaluation Against Criterion 61

3.1.2.6.2.1.1 New Fuel Storage

New fuel is placed in dry storage in the new fuel storage vault which is located inside the reactor building. The storage vault within the reactor building provides adequate shielding for radiation protection. Storage racks preclude accidental criticality (see evaluation against Criterion 62). The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

3.1.2.6.2.1.2 Spent Fuel Handling and Storage

Irradiated fuel is stored in the spent fuel pool in the reactor building. Fuel pool water is circulated through the fuel pool cooling and cleanup (FPC) system to maintain fuel pool water temperature, purity, water clarity, and water level. Storage racks preclude accidental critically (see evaluation against Criterion 62).

Reliable decay heat removal is provided by the closed loop FPC system. It consists of two circulating pumps, two heat exchangers, two filter-demineralizers, two skimmer surge tanks, and the associated piping, valves, and instrumentation. The

pool water is circulated through the system, suction is taken from surge tanks, flow passes through the heat exchanger and filters, and it is discharged through diffusers at the bottom of the fuel pool. Pool water temperature is maintained below 125°F. The FPCC system can be interconnected with the RHR system to increase the cooling capacity of the FPCC system during plant shutdown.

High and low level switches indicate pool water level changes in the main control room. Fission product concentration in the pool water is minimized by use of the filter-demineralizer. This minimizes the release of radioactivity from the pool to the reactor building environment.

No special tests are required to insure system operability because at least one pump, heat exchanger, and filter-demineralizer are continuously in operation while fuel is stored in the pool. Duplicate units are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

3.1.2.6.2.1.3 Radioactive Waste Systems

The radioactive waste systems provide all equipment necessary to collect, process, and prepare for disposal all radioactive liquids, gases, and solid waste produced as a result of reactor operation.

Liquid radwastes are segregated and treated as equipment drain, floor drain, chemical, detergent, sludges or concentrated wastes. Processing methods include filtration, ion exchange, neutralization, concentration, solidification, analysis, and dilution. Liquid wastes are also decanted and sludge is accumulated for disposal as solid radwaste. Wet solid wastes and concentrates are normally solidified and packaged in shielded steel containers. Dry solid radwastes are packaged in steel drums, or other suitable containers. Gaseous radwastes are monitored, processed, recorded and controlled so that radiation doses to persons outside the controlled area are below those allowed by applicable regulations.

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
10. RHR System (Figure 3.2-6)							
.1 Heat exchangers, primary side	GE	2	R	B	I	I	
.2 Heat exchangers, secondary side	GE	3	R	C	I	I	
.3 Piping, within outermost isolation valves, reactor coolant pressure boundary	P	1	C,R	A	I	I	
.4 Piping, other	P	2	R	B	I	I	
.5 Pumps	GE	2	R	B	I	I	
.6 Pump motors	GE	2	R	N/A	I	I	
.7 Valves, isolation, Reactor Coolant Pressure Boundary	P	1	C,R	A	I	I	
.8 Valves, other	P	2	R	B	I	I	(12)
.9 Mechanical modules	GE	2	R	B	I	I	
.10 Electrical modules with safety function	GE	2	R	N/A	I	I	
.11 Cable, with safety function	P	2	C,R,W	N/A	I	I	
11. Low Pressure Core Spray (Figure 3.2-7)							
.1 Piping, within outermost isolation valves to reactor vessel	P	1	C,R	A	I	I	(12)
.2 Piping, beyond outermost isolation valves	P	2	R	B	I	I	(12)
.3 Pumps	GE	2	R	B	I	I	
.4 Pump motors	GE	2	R	N/A	I	I	
.5 Valves, isolation, Reactor Coolant Pressure Boundary	P	1	C	A	I	I	(12)
.6 Valves, other	P	2	C,R	B	I	I	(12)
.7 Electrical modules with safety function	GE	2	R	N/A	I	I	
.8 Cable, with safety function	P	2	R,W	N/A	I	I	

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
12. High Pressure Core Spray (Figure 3.2-7)							
.1 Piping, within outermost isolation valve	P	1	C,R	A	I	I	(12)
.2 Piping, return test line to condensate storage tank beyond second isolation valve	P	G	R,O	D	II	II	(32)
.3 Piping, beyond outermost isolation valve, other	P	2	R	B	I	I	(12)
.4 Pump	GE	2	R	B	I	I	
.5 Pump motor	GE	2	R	N/A	I	I	
.6 Valves, beyond diesel shutoff valves	P	3	P	C	I	I	
.7 Valves, isolation, Reactor Coolant Pressure Boundary	P	1	C	A	I	I	
.8 Valves, beyond isolation valves, motor-operated	GE	2	R	B	I	I	(12)
.9 Valves, other	P	2	R,P	B	I	I	
.10 Electrical modules, with safety function	GE	2	R	N/A	I	I	
.11 Electrical auxiliary equipment	GE	3	DG	N/A	I	I	
.12 Cable with safety function	P	2	W,R	N/A	I	I	

(HPCS Emergency Power Supply - see 38a)

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TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
13. RCIC System (Figure 3.2-8)							
.1 Piping, within outermost isolation valves Reactor Coolant Pressure Boundary	P	1	C,R	A	I	I	(12)
.2 Piping, beyond outermost isolation valves	P	2	R	B	I	I	(12 & 23)
.3 Piping, return test line to condensate storage tank beyond second stop valve, drip pot discharge valve to condenser	P	G	R	D	II	II	(12 & 32)
.4 Pumps	GE	2	R	B	I	I	
.5 Valves, isolation and Coolant Pressure Boundary	P	1	C	A	I	I	(12)
.6 Valves, other	P	2	R	B	I	I	(13)
.7 Turbine	GE	2	R	N/A	I	I	
.8 Electrical modules, with safety function	GE	2	R	N/A	I	I	
.9 Cable, with safety function	P	2	R,W	N/A	I	I	
14. Fuel Service Equipment							
.1 Fuel preparation machine	GE	3	R	N/A	I	I	
.2 General purpose grapple	GE	3	R	N/A	I	I	
15. Reactor Vessel Service Equipment							
.1 Steam line plugs	GE	3	R	N/A	I	I	
.2 Dryer and separator sling and head strongback	GE	3	R	N/A	I	I	
16. In-Vessel Service Equipment							
.1 Control rod grapple	GE	3	C	N/A	I	I	

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
17. Refueling Equipment							
.1 Refueling equipment platform assembly	GE	3	C	N/A	I	I	
.2 Refueling Bellows	P	G	C,R	D	II	I	(33)
18. Storage Equipment							
.1 Fuel storage racks	GE/P	3	R	N/A	I	I	
.2 Defective fuel storage container	GE	3	R	N/A	I	I	
19. Radwaste System (Figures 11.2-2, 3.2-9, 3.2-10, 11.2-3, 11.2-4a thru 11.2-4c, 11.4-1a, 11.4-1b)							
.1 Tanks, Atmospheric	GE/P	G	W	C	II	II	(16, 24 & 38)
.2 Heat exchangers	GE/P	G	W	C	II	II	(16 & 24)
.3 Piping and valves forming part of containment boundary	P	2	C,R	B	I	I	
.4 Piping, other	P	G	W	C	II	II	(16 & 38)
.5 Pumps	GE/P	G	W	C	II	II	(16, 24 & 38)
.6 Valves, flow control and filter system	GE/P	G	W	C	II	II	(16, 24 & 38)
.7 Valves, other	P	G	W	C	II	II	(12, 16, 24 & 38)
.8 Mechanical modules	GE/P	G	W	C	II	II	(16 & 24)
.9 Radioactive Equipment & Floor Drains and other radwaste piping and valves upstream of collector tanks	P	G	R,T,W	D	II	II	(32)
.10 Instrumentation and control boards	GE/P	G	W	N/A	II	II	
.11 Concentrator	GE	G	W	C	II	II	
.12 Plant discharge line	GE/P	G	W	D	II	II	(37)
20. Reactor Water Cleanup System (Figure 3.2-11)							
.1 Vessels, filter/demineralizer	GE	G	W	C	II	II	
.2 Heat exchangers	GE	G	W	C	II	II	
.3 Piping, within outermost isolation valves	P	1	C	A	I	I	(12)
.4 Piping, beyond outermost containment isolation valves	P	G	R,W	C	II	II	(12 & 32)
.5 Pumps	GE	G	R	C	II	II	(12 & 32)
.6 Valves, isolation valves Reactor Coolant Pressure Boundary	P/GE	1	C,R	A	I	I	(12)

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
.7 Valves, beyond outermost containment isolation valves	GE/P	G	R,W	C	II	II	(12 & 32)
.8 Mechanical modules	GE	G	R,W	C	II	II	(32)
21. Fuel Pool Cooling and Clean-up System (Figure 3.2-12)							(18)
.1 Vessels, filter/demineralizers	P	G	W	C	II	II	
.2 Vessels, other	P	G	W	C	II	II	
.3 Heat exchangers	P	G	R	C	II	II	(32)
.4 Piping	P	G	R,W	C	II	I/II	(32)
.5 Pumps	P	G	R	C	II	II	(32)
.6 Makeup System (normal)	P	G	R	C	II	II	(18 & 32)
.7 RHR Connection, Makeup	P	3	R	C	I	I	
.8 Makeup System (emergency)	P	3	R	C	I	I	
.9 Piping, suppression pool to outer isolation valves	P	2	R	B	I	I	
22. Control Room Panels							
.1 Electrical modules with safety function	GE	2	W	N/A	I	I	
.2 Cable, with safety function	GE/P	2	W	N/A	I	I	
23. Local Panels and Racks							
.1 Electrical modules, with safety function	GE	2	R	N/A	I	I	
.2 Cable, with safety function	P	2	R	N/A	I	I	
24. Off-Gas System (Figure 11.3-2)							
.1 Tanks	GE	G	T,W	C	II	II	(16)
.2 Heat exchangers	GE	G	T,W	C	II	II	(16)
.3 Piping	P	G	T,W,O	C	II	II	(16)
.4 Pumps	GE	G	T,W	C	II	II	(16)
.5 Valves	P	G	T,W	C	II	II	(16)

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
.6 Mechanical modules, with safety function	GE	G	T,W	C	II	II	(16 & 12)
.7 Pressure vessels	GE	G	T,W	C	II	II	(16)
25. Standby Service Water System (Figure 3.2-13)							
.1 Piping	P	3	P,R,DG,O	C	I	I	
.2 Pumps	GE	3	P	C	I	I	
.3 Pump motors	GE	3	P	N/A	I	I	
.4 Valves	P	3	P,R,DG,O	C	I	I	
.5 Electrical modules, with safety function	P	3	P,R,DG,O,W	N/A	I	I	
.6 Cable, with safety function	P	3	P,R,DG,O,W	N/A	N/A	I	
26. Turbine Plant Service Water (Figure 9.2-1)							
.1 Piping and Valves	P	G	T,R,O,P,W	D	II	II	(32)
.2 Pumps	P	G	P	D	N/A	II	
27. Reactor Building Closed Cooling Water System (Figure 3.2-14)							
.1 Heat Exchangers	P	G	R	D	II	II	(32)
.2 Pumps	P	G	R	D	II	II	(32)
.3 Tanks	P	G	R	D	II	II	(32)
.4 Piping and Valves Inside Containment	P	G	C	C	II	II	(32)
.5 Containment Isolation Valves and Associated Piping	P	2	C,R	B	I	I	
.6 Piping and Valves in Reactor Building	P	G	R	D	II	II	(32)
.7 Piping and Valves Other	P	G	W	D	II	II	

TABLE 3.2-1 (Continued) Page 19 of 23

Notes (Continued)

e. Wheel and shaft forgings are ultrasonically tested according to ASTM-A-388

f. Butt-welds are radiographed according to ASME Section III, IX-3300, and magnetic particle or liquid penetrant tested according to ASME Section III, IX-3500 or IX-3600

g. Notification is made on major repairs, and records maintained thereof

h. Record system and traceability according to ASME Section III, NA-4900

i. Control and identification according to ASME Section III, NA-4400

j. Procedures conform to ASME Section III, NA-4400

k. Inspection personnel are qualified according to ASME Section III, Appendix IV, Paragraph IX-400.

14. The hydraulic control unit (HCU) is a General Electric factory-assembled engineered module of valves, tubing, piping, and stored water which controls a single control rod drive by the application of precisely timed sequences of pressure and flows to accomplish slow insertion or withdrawal of the control rods for power control, and rapid insertion for reactor scram.

Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Quality Groups A, B, C, D pressure integrity quality levels clearly apply to all levels to the interfaces between the HCU and the connecting conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components and instruments).

TABLE 3.2-1 (Continued)

Notes (Continued)

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but these codes and standards are supplemented with additional requirements for these parts and for the remaining parts and details. For example, (1) all welds are LP inspected, (2) all socket welds are inspected for gap between pipe and socket bottom, (3) all welding is performed by qualified welders, (4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer's standards and proven design techniques which are not explicitly defined within the codes of Quality Groups A, B, or C. This is supplemented by the QC techniques described above.

15. Only equipment associated with a safety action (e.g., isolation) need conform to a safety function.

16. Chapter 15 conservatively analyzes a postulated simultaneous failure of all the radwaste tanks in the radwaste building. The analysis assumes that 1 percent of the iodine is released to the atmosphere and, at the time of failure, all tanks are filled to capacity (this condition is not normally expected). The analysis evaluates the possible control room, site boundary, and low population zone exposures to the whole body and thyroid. The results of the analysis indicate, in light of the requirements of Regulatory Guide 1.29, Rev. 1, that the radwaste system is properly classified and changes are not required.

17. DELETED

18. To comply with Regulatory Guides 1.26, Rev. 3 and 1.29, Rev. 1, the RHR system is interconnected to the fuel pool, thereby providing a redundant Seismic Category I sources of coolant to the fuel pool. Additionally, systems for maintaining water quality and quantity are designed so that any malfunction or a failure in such

TABLE 3.2-1 (Continued)

Notes (Continued)

systems will not cause significant loss or inventory. In addition, a Seismic Category I makeup source which can be used as makeup as well as evaporative cooling is supplied by the standby service water system.

19. The main steam line extending from the outermost containment isolation valve up to but not including the turbine main steam stop valve, and connected piping of 2-1/2 inches or larger nominal pipe size up to and including the first isolation valve is designed by use of an appropriate seismic system analysis for the SSE and OBE. The power conversion system structures are constructed in accordance with applicable codes for steam power plants. The turbine building, interacting with main steam lines and branch lines, is designed as a modified non-Category I Seismic structure as described in 3.8.4.1.3.

20. The condensate storage tanks are designed, fabricated, and tested to meet ASME Code, Section III, subsection ND-3800. In addition, the specification for this tank requires 100 percent surface examination of the side wall to bottom joint, and 100 percent volumetric examination of the side wall weld joints.

21. Not Used.

22. These lines meet the requirements of Quality Group B except that hydrostatic testing of the containment spray piping is not required.

23. The RCIC turbine exhaust line from the isolation valve to the suppression pool meets all the requirements of Quality Group B except that hydrostatic testing of this portion of piping is not required.

24. Equipment, piping, and valves which are part of the radwaste system but not used for processing radioactive fluids are designed to Quality Group D standards.

Notes (Continued)

25. Not used.

26. Portions of the turbine building that support or interact with main steam piping are designed to Seismic Category I.

27. Those portions of the radwaste and control building that house systems or components necessary for safe shutdown of the reactor are designed to Quality Class I and Seismic Category I requirements. Those portions of the radwaste building housing equipment containing significant quantities of radioactive material are designed to Seismic Category I requirements.

28. Lavatory exhaust systems are designed to Quality Assurance Class G.

29. Nonessential equipment and components are designed to Seismic Category II and Quality Assurance Class II.

30. The high pressure core spray suction piping from the condensate storage tank provides the initial source of makeup water to the high pressure core spray system for safety injection. Consequently, this piping has been upgraded by full volumetric examination of every weld.

31. The makeup water pumphouse is designed to withstand the Design Basis Tornado. The design also considers the possible effects of tornado generated missiles. The tower makeup water piping, valves and cabling located underground are provided with adequate earth cover to be resistant to tornado generated missiles or are protected by tornado-resistant structures.

32. Nonessential piping systems, HVAC, cable trays, and system components in the reactor building, primary containment, the control building, diesel generator building and the standby service water pumphouses are supported as Seismic Category I systems regardless of service. All hangers and supports in these systems are fabricated and installed in accordance with Quality Class I requirements.

3.4.1.4.1.2 Internal Flood Protection Requirements

Section 3.4.1.5.2 discusses internal flood protection measures provided for safety-related systems, equipment and components.

Figures 1.2-3, 1.2-6, 3.8-1, 3.8-2 and 3.8-45 illustrate plant arrangement and layout and show that safety-related equipment is located within individual rooms or compartments. The reinforced concrete walls and watertight doors between the rooms or compartments act as positive barriers against possible means of flooding.

The potential flooding attributable to, and the flooding and environmental effects of, postulated through-wall leakage cracks in moderate energy fluid piping systems, and postulated rupture of high energy fluid piping systems are evaluated. These are discussed in 3.6.1.

Section 6.3 addresses single failure of ECCS piping, including leak detection requirements for ECCS passive failures, ECCS passive failures during long-term cooling, and potential flooding attributable thereto.

Section 9.3.3 discusses the design bases, system decriptions, safety evaluation, testing and inspection requirements, and the instrumentation requirements relative to equipment and floor drainage systems. The design bases used ensure equipment and floor drainage systems integrity during normal plant operation and preclude any danger to health and safety of plant personnel, the environs and the general public. Five independent sumps are provided in the reactor building at floor elevation 422'-3" as shown on Figures 1.2-6 and 1.2-7. One sump is the equipment drain sump for the reactor building (see Figure 9.3-5). The other four sumps are floor drain sumps serving the watertight pump rooms which surround the primary containment of the basement level (elevation 422'-3"). The sumps collect water from typical sources as equipment drains from equipment carrying low purity water and from floor and pit drains. In the event of a pipe break of sufficient size to flood sump pumps in one room, common mode flooding between watertight rooms is prevented by the following:

a. Of the safety-related watertight pump rooms at the 422'-3" level of the Reactor Building the equipment drain sump serves only the RCIC pump room. Drains to RHR A and B pump rooms are capped. See 9.3.3.2.1 for more detail.

b. The floor drain sumps serving more than one pump room have two isolation valves in the drain headers from adjacent pump rooms which automatically close on a high sump level. Accordingly, floor drainage in any pump room exceeding the capacity of the sump pump will be confined to the room in which the leakage occurs. The high sump level also annunciates in the Main Control Room. See 9.3.3.2.2.1 for more details.

c. Wall-mounted Class 1E level switches are also located in each pump room just above floor level. These alarms ensure that if a failure in the sump isolation and alarm system should occur in any of the rooms, prompt operator notification of the problem would be received in time to isolate any leak. See 6.3.2.5 and 9.3.3.2.2.1 for more details.

Administrative controls assure that separation criteria is maintained and watertight doors and hatches are closed as appropriate.

TABLE 3.7-2 Page 1 of 3

LUMPED REPRESENTATION OF SOIL-STRUCTURE INTERACTION

A. For a Rectangular Foundation

Motion	Equivalent Spring Constant Note 2	Equivalent Damping Coefficient Notes 2,3
Vertical	$k_v = \frac{G}{1-\nu} \beta_v \sqrt{4cd}$	See Note 1
Horizontal	$k_h = 4(1+\nu) G\beta_h \sqrt{cd}$	See Note 1
Rocking	$k_r = \frac{G}{1-\nu} \quad \beta_R \cdot 8cd^2$	See Note 1
Torsional	See Note 1	See Note 1

B. For a Circular Foundation

Motion	Equivalent Spring Constant Note 2	Equivalent Damping Coefficient Notes 2,3
Vertical	$k_v = \frac{4GR}{1-\nu}$	$c_v = 0.85k_v \, R\sqrt{\rho/G}$
Horizontal	$k_h = \frac{32(1-\nu)GR}{7-8\nu}$	$c_h = 0.575k_h \, R\sqrt{\rho/G}$
Rocking	$k_R = \frac{8GR^3}{3(1-\nu)}$	$c_R = \frac{0.30}{1+B_R} k_R \, R\sqrt{\rho/G}$
Torsional	$k_t = \frac{16GR^3}{3}$	$c_R = \frac{\sqrt{k_t \, I_t}}{1 + 2B_t}$

TABLE 3.7-2 (Continued) Page 2 of 3

NOTES:

1. Use formulas and diagrams for an equivalent circular base with a radius determined by the following:

For translation: $R = \sqrt{\frac{4cd}{\pi}}$

For rocking: $R = \sqrt{\frac{16\ cd^3}{3\pi}}$

For torsion: $R = \sqrt[4]{\frac{16cd\ (c^2 + d^2)}{6\pi}}$

2. In above formulas:

$2c$ = width of the rectangular foundation (along axis of rotation for the case of rocking);

$2d$ = length of the rectangular foundation (in the plane of rotation for rocking);

β_v, β_h, and β_R = constants depending on the ratio d/c (See Figure 10-16, p. 351 of Reference 5)

R = radius of the circular foundation

$$B = \frac{3(1-\nu)\ I_R}{8\rho R^5}$$

I_R = total mass moment of inertia of structure and foundation about the rocking axis at the base

$$B_T = {}^{I_T}/_{\rho}R^5$$

accident condition tests include exposure to steam and containment water spray solutions under temperature-time conditions which are more severe than those that would be encountered in a design basis accident.

All exterior surfaces of the steel primary containment vessel shell are cleaned in accordance with the requirements of Steel Structures Painting Council Surface Preparation Specification No. 6, Commercial Blast Cleaning, latest edition. After cleaning and after having passed inspection, one prime coat of Amercoat Corporation Dimetcote No. 6, minimum 3.0 mils thick, is applied.

All interior surfaces of the primary containment vessel shell and metal surfaces of attachments thereto, except those parts embedded in the base slab, are given the following protective coatings:

a. The drywell: one prime coat of Ameron Corporation Dimetcote No. 6 topcoated with one coat of Amercoat 90 modified phenolic epoxy coating. Surfaces receive an SP-10 surface preparation.

b. The non-immersion surfaces of the suppression chamber: one coat of Dimetcote No. 6. Surfaces receive an SP-10 surface preparation.

c. The immersion surfaces of the suppression chamber: two coats of Amercoat 90, 10 to 14 mil dry film thickness. Immersion surfaces receive an SP-5 surface preparation.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria for the steel primary containment vessel, namely, the basis for establishing allowable stress values, the deformation limits, and the factors of safety, are established by and in accordance with the ASME Code, Section III.

In addition to the structural acceptance criteria, the steel primary containment vessel is designed to meet minimum leakage rate requirements. The leakage rate requirements are discussed in Chapter 6.

3.8.2.5.1 Stress Limits for Design Loading Conditions

The requirements of Section 3000 of the ASME Code, Section III, as modified by Regulatory Guide 1.57, Rev. 0 and discussed in 3.8.2.2.4.6, are met for each of the load

tions prescribed in 3.8.2.3.12, Loading Combinations. This is discussed in greater detail in 3.8.2.4, Design and Analysis Procedures. Buckling criteria is discussed in 3.8.2.5.4, Buckling Criteria for the Primary Containment Vessel.

3.8.2.5.2 Primary and Secondary Stresses

For loading combinations described in 3.8.2.3.12a through 3.8.2.3.12e, inclusive and 3.8.2.3.12g, the stress limits specified in ASME Code, Section III, NE 3131 (c) are utilized.

3.8.2.5.3 Peak Stresses

For loading combinations described in 3.8.2.3.12f and 3.8.2.3.12h, the stress limits specified in the ASME Code, Section III, NE 3131(c) (1) and NE 3131(c) (2) are utilized.

3.8.2.5.4 Buckling Criteria for the Primary Containment Vessel

To assure safety against buckling, the rules set forth in the ASME Code, Section III, NE 3133 are utilized.

The buckling analysis of the containment vessel was performed as follows:

External Pressure - The allowable working pressure, Pa, calculated in NE-3133.3 was compared with the specified maximum external pressure, -4 psi. Conical shell elements were analyzed as equivalent cylinders in accordance with NE-3133.7.

Longitudinal Compression on Unstiffened Shell - The maximum allowable compressive stress, B, determined in NE-3133.6 was compared to the maximum longitudinal compressive stress produced under all the loading conditions specified, including the compressive stress due to the SSE overturning moment.

Longitudinal Compression Meridionally Stiffened Shell - Two independent checks were made on buckling of stiffened shell lengths:

a. NE-3133.6 was applied as above using an equivalent thickness in bending, $t_e = (12 \times I_s/b)^{1/3}$

   b = meridional stiffener spacing

   I_s = moment of inertia of the composite section comprised of the stiffener and a width of shell, b.

3.8.4 OTHER SEISMIC CATEGORY I STRUCTURES, AND NON-SEISMIC CATEGORY I SAFETY RELATED STRUCTURES

3.8.4.1 Description of Structures

The following provides descriptive information of the various structures, other than the primary containment vessel and its internal structures, to define their primary structural aspects and elements relied upon to perform their safety related functions. The relation between adjacent structures, including the separations provided, is also discussed. Figures 1.2-1 to 1.2-14, inclusive and Figures 3.8-1, 3.8-2, 3.8-30 to 3.8-44 and 3.8-46, inclusive show arrangements and details of these structures.

The various plant structures discussed are the following:

a. Reactor building

b. Radwaste and control building

c. Turbine generator building

d. Diesel generator building

e. Spray ponds and standby service water pumphouses

f. Makeup water pumphouse and associated structures of the cooling tower makeup water system, such as valve box structures at points along the makeup water underground pipeline

g. Condensate storage tanks retaining area

h. Fresh air intake structures

The reactor building, radwaste and control building, turbine generator building, diesel generator building and the service building are grouped together to form a plant complex. However the buildings are separated from each other by gaps, and are supported on separate foundation mats.

The safety related functions fulfilled by each of the above structures are discussed in subsections of 3.8.4.1 that follow.

3.8.4.1.1 Reactor Building

The reactor building is located within the interior of the plant complex. The general arrangement and principal features of the reactor building are shown in Figures 1.2-1 to 1.2-7, inclusive, 3.8-1, 3.8-2 and 3.8-30 through 3.8-45, and 3.8-47, 3.8-54, 3.8-55, 3.8-60 and 3.8-61 inclusive.

The reactor building is part of the secondary containment system. The reactor building completely encloses the reactor vessel and the primary steel containment vessel and provides secondary containment when the primary containment vessel is sealed and in service. When the primary containment vessel is open, as it is during refueling periods, the reactor building also provides primary containment. The building houses the primary reactor system, reactor auxiliary and cooling systems, and facilities necessary for refueling operations.

The reactor building, as the secondary containment structure, houses safety related and other systems, equipment and components which include:

a. Refueling and reactor servicing equipment

b. New and spent fuel storage facilities

c. Other reactor auxiliary and service equipment, including:

   (1) Reactor core isolation cooling system

   (2) Reactor water cleanup system

   (3) Standby liquid control system

   (4) Control rod drive system equipment

   (5) Emergency core cooling system

   (6) Electrical equipment and components

   (7) Supply and exhaust air ventilating system

   (8) Standby gas treatment system (SGTS) equipment

g. Cable room air handling units (2)

h. Control room air handling units (2)

i. RHR pump room cooling coil (3)

j. RCIC pump room cooling coil

k. LPCS pump room cooling coil

l. HPCS pump room cooling coil

m. Motor control center room cooling coils (5)

n. Diesel generator room cooling coils (7)

o. Standby service water pumphouse (2)

To cool the items of equipment listed above, the standby service water pumps take suction from the spray ponds and pump water through the various heat exchangers and coolers required for normal and emergency shutdown. The water is returned to the spray ponds through the spray distribution piping shown in Figure 1.2-14. Emergency power is provided to all equipment required for operation of the standby service water system.

The standby service water system is classified as Quality Group C as defined in Regulatory Guide 1.26, Rev. 3, and in Table 3.2-1.

The standby service water system is an ASME Section III, Class 3 system, and is designed, fabricated and constructed in accordance with Quality Class I requirements of the Quality Assurance Program.

The standby service water system, including the spray pond structures 1A and 1B and the pumphouse structures 1A and 1B, are designed to Seismic Category I requirements as defined in Regulatory Guide 1.29, Rev. 1, and in Table 3.2-1.

The spray distribution systems in each spray pond are completely redundant, and each spray distribution system is capable of providing sufficient cooling to safely shutdown the plant.

The spray headers need not be protected from tornadoes and tornado missiles since a simultaneous tornado and LOCA is not a design basis event. In the event of damage to the spray distribution piping, the cooling tower makeup water system, which is impervious to tornado damage, is available to pump water at a maximum temperature of 70°F to the spray ponds.

The ability of the spray ponds and associated spray distribution systems, piping and pipe supports, and the standby service water pumphouses to withstand the wind effects associated with the design basis tornado and tornado induced missiles is addressed in 3.3. Based on the discussion in 3.3, the ultimate heat sink has sufficient capability to perform its safety related function in response to a postulated tornado event.

As discussed in 3.4, the design basis flood elevation is 433'-0" feet mean sea level (MSL) which includes wind wave action. The plant site elevation is 441 feet above MSL, except at the spray ponds where the finish grade is 434 feet above MSL. These elevations are sufficient to protect the plant site, including the spray ponds, against flood conditions. Since the spray ponds and pumphouses are located at sufficient elevation and distance from the Columbia River, the effects of the design basis flood including wind wave action and spray do not require flood protection measures.

The spray ponds and pumphouses are located above the present groundwater elevation of 380 feet above MSL and are not subjected to any force effects of bouyancy and static water from this groundwater elevation. Uplift and lateral hydrostatic pressure are considered, to ensure the safety of the spray ponds and pumphouses in the event of a rise in the groundwater table to the postulated elevation of 420 feet above MSL should construction of Ben Franklin Dam be realized, as discussed in 3.4.

The spray pond structures are reinforced concrete rectangular shaped retention basins consisting essentially of a structural slab on soil and four perimetral exterior walls. The walls and floor slab are bounded by Quality Class I high relative density backfill. Each pond is approximately 250 feet square

TABLE 3.8-9 (Continued)

REFERENCE NUMBER	DESIGNATION	TITLE	EDITION
24	ASME	1971 ASME Boiler & Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Reactor Coolant Systems"	Summer of 1972 Addenda
25	ASTM	Annual Books of ASTM Standards	1972
26	ANSI B31.1.0	Standard Code for Pressure Piping, Power Piping	Latest Edition
27	API Spec. No. 620	Specification for Welded Steel Storage Tanks	Feb. 1970
28	UBC	Uniform Building Code	1970
29	NEC	National Electric Code	Latest Edition
30	CRD-C-588	U.S. Corps. of Engineers Specification for Expansive Grouts	Latest Edition
31	CRD-C-589	U.S. Corp. of Engineers Methods of Sampling and Testing Expansive Grouts	Latest Edition
32	CRSI	Manual of Standard Practice	1972

TABLE 3.8-9 (Continued) Page 5 of 7

REFERENCE NUMBER	DESIGNATION	TITLE	EDITION
33	ANSI 45.2.5-74	Supplementary Quality Assurance Requirements for Installations, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1974
34	----	Steiger Occupational Safety and Health Act	Latest Edition
35	NRC Regulatory Guide 1.10	Mechanical Cadweld Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1)	Jan. 2, 1973
36	NRC Regulatory Guide 1.12	Instrumentation for Earthquakes (Revision 1)	April, 1974
37	NRC Regulatory Guide 1.13	Fuel Storage Facility Design Basis	March 10, 1971
38	NRC Regulatory Guide 1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1)	Dec. 28, 1972
39			
40	NRC Regulatory Guide 1.26	Quality Group Classification and Standards (Revision 3)	Sept. 1974
41	NRC Regulatory Guide 1.27	Ultimate Heat Sink for Nuclear Power Plants (Revision 1)	March 1974
42	NRC Regulatory Guide 1.29	Seismic Design Classition (Revision 1)	August 1973
43	NRC Regulatory Guide 1.31	Control of Stainless Steel Welding (Revision 1)	June 1973

The heat exchanger, including its appurtenances and supports, is designed to withstand the static seismic accelerations specified below. The seismic coefficients are applied at the center of gravity of the heat exchanger, assuming the heat exchanger to be flooded.

Seismic Acceleration Coefficients for the RHR heat exchanger are:

(OBE)	1.5g horizontal 0.14g vertical
(SSE)	3.0g horizontal 0.28g vertical

3.9.2.2.2.12 Standby Liquid Control Tank

The standby liquid control storage tank is a cylindrical tank 9 feet in diameter and 12 feet high bolted to the concrete floor. Stresses can be calculated readily by conventional methods. The magnitude of the earthquake coefficients for Safe Shutdown Earthquake (SSE) are 3.0g horizontal and 0.29g vertical. The Standby Liquid Control Tank has been qualified by analysis for:

a. Stresses in the tank bearing plate

b. Belt stresses

c. Sloshing loads imposed by earthquake natural frequency of sloshing = 0.58 Hz

d. Minimum wall thickness

e. Buckling

3.9.2.2.2.13 Main Steam Isolation Valves

The main steam isolation valve structures have been analyzed and representative models statically tested to demonstrate operability at the specified SSE accelerations. Static testing consisted of mechanically loading the extended mass of the valve actuator to equivalent seismic loading while valve closure was performed. Operation of the valve was demonstrated during this test.

3.9.2.2.2.14 Main Steam Safety/Relief Valves

Due to the complexity of this structure and the performance requirements of the valve, the total assembly of the safety/relief valve (including electrical, pneumatic devices) was dynamically tested at seismic accelerations equal to or greater than the SSE levels determined for this plant. Satisfactory operation of the valves was demonstrated during and after the test.

3.9.2.2.2.15 Fuel Pool Cooling and Cleanup and Motor Assembly

The fuel pool cooling and cleanup system pump and motor have not been analyzed as Seismic Category I equipment since this was not a requirement of the construction permit.

3.9.2.2.2.16 Balance of Plant Safety-Related Mechanical Equipment

Balance of plant Seismic Category I equipment, components and accessories are designed based on results determined analytically (see 3.9.2.2) or through dynamic testing. The dynamic program is performed to confirm the ability of the equipment to function as needed during and after an earthquake of magnitude up to and including the SSE. These test programs implement the criteria stated in 3.9.2.2, 3.9.2.2.1, 3.9.2.2.1.1, 3.9.2.2.1.2, 3.9.2.2.1.3, and 3.9.2.2.1.4. The dynamic tests met the seismic loading requirements as defined by the applicable floor response spectrum curves for the appropriate damping coefficients.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady State Conditions

The major reactor internal components within the vessel were subjected to extensive testing coupled with dynamic system analyses to properly describe the resulting flow-induced vibration phenomena incurred from normal reactor operation and from anticipated operational transients.

In general, the vibration forcing functions for operational flow transients and steady state conditions are not predetermined by detailed analysis. Special analyses of the response signals measured from reactor internals of similar

The design loading combinations and limits for the pump include the following:

a. Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, dead weight loads including seismic due to operational basis earthquake (OBE) loads, plus torsional loads due to rotation of the component assembly.

b. Seismic loading: This equipment and supports are designed to withstand the static seismic forces applied at the mass center, assuming that the pump is flooded.

c. Stresses in the supports and the anchor bolts due to seismic loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the applicable codes.

d. The ASME Boiler and Pressure Vessel Code, Section III, is used as a guide in calculating the thickness of the pressure retaining parts and for sizing the cover bolting.

e. Identified thermal transients: Equipment operates between 70 - 545°F. Transient analysis is not required for Class III components in this temperature range.

Table 3.9-2(p) shows the calculated stress values and allowable stress limits for the pump.

3.9.3.1.16 Fuel Pool Cooling and Cleanup System Heat Exchangers and Pumps

The fuel pool cooling and cleanup heat exchangers and pumps are not part of a safety related system. They are designed to the requirements of Seismic Category II as described in 3.2.1.

3.9.3.1.17 Reactor Water Cleanup System (RWCU) Heat Exchangers

The RWCU regenerative and non-regenerative heat exchangers are not part of a safety system and are not designed to Seismic I requirements.

No experimental or inelastic stress analysis was used in the design of these heat exchangers.

The loading considered in the design of the heat exchangers include:

a. Normal plus upset loads: This includes the simultaneous effect of normal operating loads, design pressure, temperature, nozzle loads, dead weight loads.

b. Seismic loading: The equipment and supports are designed to withstand the static seismic forces applied.

c. Stresses in the supports and the anchor bolts due to seismic loads are combined with the stresses due to other live and dead loads and operating loads. The allowable stress for this combination of loads is based on the allowable stress as set forth in the applicable codes.

d. The allowable shear on anchor bolts set in concrete are in accordance with Table No. 26-1 of the Uniform Building Code.

Table 3.9-2(c) shows the calculated stress values and allowable stress limits for the heat exchangers.

3.9.3.1.18 Control Rod Drive Piping

The safety related portion of the control rod drive (CRD) piping is designed in accordance with the ASME Code, Section II, Subsection NB for Class 1 piping and NC for Class 2 piping. The load combinations and activities are shown in Table 3.9-16 for Class 1 piping and Table 3.9-17 for Class 2 piping.

The remainder of the CRD piping, which is not safety related, is designed in accordance with ANSI B31.1.

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e. Metal piston rings are Haynes 25 alloy.

f. Certain wear surfaces are hard-faced with Colmonoy 6.

g. Nitriding by a proprietary new Malcomizing process and chromium plating are used in certain areas where resistance to abrasion is necessary.

h. The drive piston head is made of Armco 17-4Ph.

Pressure-containing portions of the drives are designed and fabricated in accordance with requirements of Section III of the ASME Boiler and Pressure Vessel Code.

4.6.1.1.2.4 Control Rod Drive Hydraulic System

The control rod drive hydraulic system (Figures 4.6-5a, b) supplies and controls the pressure and flow to and from the drives through hydraulic control units (HCU). The water discharged from the drives during a scram flows through the HCUs to the scram discharge volume. The water discharged from a drive during a normal control rod positioning operation flows through the HCU, the exhaust header, and into the cooling water header. There are as many HCUs as the number of control rod drives.

4.6.1.1.2.4.1 Hydraulic Requirements

The CRD hydraulic system design is shown in Figures 4.6-5a, b, and 4.6-6. The hydraulic requirements, identified by the function they perform, are as follows:

a. An accumulator hydraulic charging pressure of approximately 1400 to 1500 psig is required. Flow to the accumulators is required only during scram reset or system startup.

b. Drive pressure of approximately 260 psi above reactor vessel pressure is required. A flow rate of approximately 4 gpm to insert a control rod and 2 gpm to withdraw a control rod is required.

c. Cooling water to the drives is required at approximately 20 psi above reactor vessel pressure and at a flow rate of 0.20 to 0.34 gpm per drive unit. (Cooling water to a drive can be interrupted for short periods without damaging the drive.)

d. The drive header pressure will be no more than 5 psi above the cooling water header pressure. (The pressure in this line must be kept as low as possible to avoid interference with normal drive movement.)

e. The scram discharge volume is sized to receive and contain all the water discharged by the drives during a scram; a minimum volume of 3.34 gal. per drive is required.

4.6.1.1.2.4.2 System Description

The CRD hydraulic systems provide the required functions with the pumps, filter, valves, instrumentation, and piping shown in Figures 4.6-5a, b and described in the following paragraphs.

Duplicate components are included, where necessary, to assure continuous system operation if an in-service component requires maintenance.

4.6.1.1.2.4.2.1 Supply Pump

One supply pump pressurizes the system with water from a condensate supply header which takes suction from the condensate treatment system and/or condensate storage tanks depending on plant operation. One spare pump is provided for standby. A discharge check valve prevents backflow through the nonoperating pump. A portion of the pump discharge flow is diverted through a minimum flow bypass line to the condensate storage tank. This flow is controlled by an orifice and is sufficient to prevent immediate pump damage if the pump discharge is inadvertently closed.

Condensate water is processed by two filters in the system. The pump suction filter is a disposable element type with a 25-micron absolute rating. A 250-micron strainer in the filter bypass line protects the pump when the filter is being serviced. The drive water filter downstream of the pump is a cleanable element type with a 50-micron absolute rating. A differential pressure indicator and control room alarm monitor the filter element as it collects foreign materials.

4.6.1.1.2.4.2.2 Accumulator Charging Pressure

Accumulator charging pressure is established by the discharge pressure of the system supply pump. During scram the scram inlet (and outlet) valves open and permit the stored energy in the accumulators to discharge into the drives. The resulting pressure decrease in the charging water header allows the CRD supply pump to "run out" (i.e., flow rate to increase substantially) into the control rod drives via the charging water header. The flow sensing system upstream of the accumulator charging header detects high flow and closes the flow control valve. This action maintains increased flow through the charging water header.

Pressure in the charging header is monitored in the control room with a pressure indicator and high pressure alarm.

During normal operation the flow control valve maintains a constant system flow rate. This flow is used for drive flow, drive cooling, and system stability.

4.6.1.1.2.4.2.3 Drive Water Pressure

Drive water pressure required in the drive header is maintained by the drive pressure control valve, which is manually adjusted from the control room. A flow rate of approximately 6 gpm (the sum of the flow rate required to insert and withdraw a control rod) normally passes from the drive water pressure stage through two solenoid operated stabilizing valves (arranged in parallel) and then goes into the cooling water header. The flow through one stabilizing valve equals the drive insert flow; that of the other stabilizing valve equals the drive withdrawal flow. When operating a drive, the required flow is diverted to that drive by closing the appropriate stabilizing valve. Thus, flow through the drive pressure control valve is always constant.

Flow indicators in the drive water header and in the line downstream from the stabilizing valves allow the flow rate through the stabilizing valves to be adjusted when necessary. Differential pressure between the reactor vessel and the drive pressure stage is indicated in the control room.

4.6.1.1.2.4.2.4 Cooling Water Header

The cooling water header is located just upstream of the return line control valve. An automatic pressure regulating valve controls the pressure in this header, which is set to produce the desired cooling water flow to the drives. A flow indicator in the control room monitors cooling water flow. A differential pressure indicator in the control room indicates the difference between reactor vessel pressure and drive cooling water pressure. Although the drives can function without cooling water, seal life is shortened by long term exposure to reactor temperatures. The temperature of each drive is recorded in the control room, and excessive temperatures are annunciated.

4.6.1.1.2.4.2.5 Return Line

The return line routes excess flow from the control rod drive hydraulic system (water not used for charging of accumulators, movement of drives or cooling) through the reactor water cleanup system to the reactor feedwater line. A pressure control valve in this line is manually adjusted from the control room to produce the desired pressure. The flow through this valve is virtually constant. Therefore, once adjusted, the drive pressure control valve and the return water control valve can maintain their required pressure independent of reactor pressure.

4.6.1.1.2.4.2.6 Scram Discharge Volume

The scram discharge volume consists of header piping which connects to each HCU and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

During normal plant operation the scram discharge volume is empty, and vented to atmosphere through its open vent and drain valve. When a scram occurs, upon a signal from the safety circuit these vent and drain valves are closed to conserve reactor water. Lights in the control room indicate the position of these valves.

If the plug were to blow out while the drive was latched, there would be no control rod motion. No pressure differential would exist across the collet piston to unlatch the collet. As in the previous failure, reactor water would flow past the velocity limiter, down the annulus between the drive and thermal sleeve, through the vessel ports and drilled passage, through the ball check valve cage and out the open plug hole to the drywell. The leakage calculations indicate the flow rate would be 350 gpm. This calculation assumes liquid flow, but flashing of the hot reactor water to steam would reduce this rate to a lower value. Drive temperature would rapidly increase and initiate an alarm in the control room.

If the plug failure were to occur during control rod withdrawal, (it would not be possible to unlatch the drive after such a failure) the collet would relatch at the first locking groove. If the collet were to stick, calculations indicate the control rod withdrawal speed would be 11.8 feet per second. There would be a large retarding force exerted by the velocity limiter due to a 35 psi pressure differential across the velocity limiter piston.

4.6.2.3.2.2.8 Drive Pressure Control Valve Closure (Reactor Pressure, 0 psig)

The pressure to move a drive is generated by the pressure drop of practically the full system flow through the drive pressure control valve. This valve is either a motor operated valve or a standby manual valve; either one is adjusted to a fixed opening. The normal pressure drop across this valve develops a pressure 260 psi in excess of reactor pressure.

If the flow through the drive pressure control valve were to be stopped, as by a valve closure or flow blockage, the drive pressure would increase to the shutoff pressure of the supply pump. The occurrence of this conditon during withdrawal of a drive at zero vessel pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations indicate that the drive would accelerate from 3 in./sec to approximately 6.5 in./sec. A pressure differential of 1670 psi across the collet piston would hold the collet unlatched. Flow would be upward, past the velocity limiter piston, but retarding force would be negligible. Rod movement would stop as soon as the driving signal was removed.

4.6.2.3.2.2.9 Ball Check Valve Fails to Close Passage to Vessel Ports

Should the ball check valve sealing the passage to the vessel ports be dislodged and prevented from reseating following the insert portion of a drive withdrawal sequence, water below the drive piston would return to the reactor through the vessel ports and the annulus between the drive and the housing rather than through the speed control valve. Because the flow resistance of this return path would be lower than normal, the calculated withdrawal speed would be 2 ft/sec. During withdrawal, differential pressure across the collet piston would be approximately 40 psi. Therefore, the collet would tend to latch and would have to stick open before continuous withdrawal at 2 ft/sec, could occur. Water would flow upward past the velocity limiter piston, generating a small retarding force of approximately 120 pounds.

4.6.2.3.2.2.10 Hydraulic Control Unit Valve Failures

Various failures of the valves in the HCU can be postulated, but none could produce differential pressures approaching those described in the preceding paragraphs and none alone could produce a high velocity withdrawal. Leakage through either one or both of the scram valves produces a pressure that tends to insert the control rod rather than to withdraw it. If the pressure in the scram discharge volume should exceed reactor pressure following a scram, a check valve in the l ne to the scram discharge header prevents this pressure from operating the drive mechanisms.

4.6.2.3.2.2.11 Collet Fingers Fail to Latch

The failure is presumed to occur when the drive withdraw signal is removed. If the collet fails to latch, the drive continues to withdraw at a fraction of the normal speed. This assumption is made because there is no known means for the collet fingers to become unlocked without some initiating signal. Because the collet fingers will not cam open under a load, accidental application of a down signal does not unlock them. (The drive must be given a short insert signal to unload the fingers and cam them open before the collet can be driven to the unlock position.) If the drive withdrawal valve fails to close following a rod withdrawal, the collet would remain open and the drive continue to move at a reduced speed.

MPL NO. C12-1020

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	63	0	0	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	PR	PR	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	4	6	0	0	0		.34 MAX	.34 MAX	.34 MAX	.34 MAX	0	0	0	0	
PRESSURE PSIG	PR + 30	PR + 30	1455				PR + 15	PR + 14	PR + 14	PR	PR		PR	0	

CONDITIONS:
1. DRIVES LATCHED
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES. MINIMUM REQUIRED PRESSURE AT POSITION 1A IS 20 FEET OF WATER AT 200 GPM.

MODE A SIZES THE COOLING WATER HEADERS.

LINE LOSS FROM LOCATION 10 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

(FOR NOTES SEE SHEET 2.)

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	59	0	.7	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	<PR + 8	<PR + 8	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	2	0	4	4		0	4	4	1.3	.7	.7	.7	0	
PRESSURE PSIG	PR + 30	PR + 30	1455	PR + 260	PR + 250		PR + 15	PR + 91	PR + 90	PR	PR + 20 MAX	PR + 20 MAX	PR + 8 MAX	0	

CONDITIONS:
1. DRIVE INSERTING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS.

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	45	45	45	20	25	25	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	21	1550	1550									SEE NOTE 9	SEE NOTE 9	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	0	0	0		0	90	90	-3.6	30	30	0.1 + SEE NOTE 9	APPROX 5565	
PRESSURE PSIG								1167 MIN	731 MIN	PR	256 MAX	94		65 MAX	
														SEE NOTE 10	

CONDITIONS:
1. DRIVES SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES.

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	200	200	200	20	180	180	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	19	1210						>PR	>PR	>PR	>PR	>PR	>PR	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	155	0	0		0	0.92	0.92	0.92	SEE NOTE 9	SEE NOTE 9	0.1	0	
PRESSURE PSIG			988					76	76	PR	65 MAX	65 MAX		65 MAX	
											SEE NOTE 10				

CONDITIONS:
1. SCRAMMING OF DRIVES COMPLETED
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE.

NOTE: MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565 PSIG.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE SYSTEM (PROCESS DIAGRAM)	FIGURE 4.6-6b

TABLE I

LOCATION	1A-1B	1B--1	2---6	3A-3B	6--9	7A-7B	7B-7C
DESIGN PRESS. (PSIG.)	150	150	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	4	4	2	1	1.5	1	0.75

LOCATION	10-20	11-12	15B-15C	15-15B	16-16A	17-18	12-26
DESIGN PRESS. (PSIG.)	1750	1750	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	2**	1	0.75	1	2	1	1

LOCATION	21-22	24-25	27A-27B	27B-27	27-27C	27C-27D	5A
DESIGN PRESS. (PSIG.)	1750	1750	1250	1250	1250	1250	1750
DESIGN TEMP. (DEG F)	150	550	280	280	280	280	150
ESTIMATED LINE SIZE (INCHES)	1	0.75	0.75	*	10	2	.75

* SEE CRD SYSTEM DESIGN SPECIFICATION.
** 2 INCH HEADER TO EACH HALF OF THE TOTAL QUANTITY OF HCU'S.

NOTES:

1. DEFINITION OF SYMBOLS
   PR - INDICATES PRESSURE OF THE REACTOR
2. MAXIMUM OPERATING TEMPERATURES
   THE MAXIMUM SYSTEM OPERATING TEMPERATURE WILL NOT EXCEED 150 DEG. F. FROM LOCATION 1 THROUGH 27 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAXIMUM TEMP. (DEG. F.)
MODE A -	23	200
MODE C -	23	546
	24	546
	25	280
	27	280
MODE D -	23	200
	24	280
	25	280
	27	280

3. MODE A -
   A. MAXIMUM CHARGING WATER PRESSURE SHALL BE 1600 PSIG NOMINAL. ACCUMULATOR PRECHARGE PRESSURE SHALL BE 575 PSIG NOMINAL, 580 PSIG MAXIMUM, AT 70° F.
   B. DELETED
   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR + 15 PSIG. FOR THE CONDITIONS INDICATED.
   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.
4. MODE B -
   A. LOCATION 13 AND 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSES ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 18 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR + 250 PSIG. FOR THE CONDITIONS INDICATED.
5. MODE C -
   A. DELETED
   B. THE 546 DEG. F. TEMPERATURE LISTED IN NOTE 2 FOR MODE C POSITIONS 23 AND 24 SHALL BE USED ONLY IN DETERMINING THE MINIMUM PIPE WALL THICKNESS IN VICINITY OF THE DRIVE HOUSING AND NOT IN DETERMINING STRESSES DUE TO THERMAL EXPANSION. IN DETERMINING MINIMUM WALL THICKNESS IT MAY BE ASSUMED THAT THIS TEMPERATURE OCCURS LESS THAN 1 PERCENT OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYD. SYSTEM DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESSES DUE TO THERMAL EXPANSION.
   C. LOCATION 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 435 PSI AT 90 GPM FOR ANY CRD.
   D. LOCATION 23 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL, INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE THE REACTOR CAN REACH 10 GPM PER DRIVE.
   E. LOCATION 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 30 GPM FOR ANY CRD.
   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.
   G. SCRAM DRAIN VALVE F011 AND VENT VALVE F010 CLOSE WITH A SCRAM SIGNAL.
6. MODE D -
   A. DELETED
   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 100 PERCENT STROKE IS LESS THAN 65 PSIG.
7. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.
8. PROCESS DIAGRAM 112D1448 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.
9. DURING SCRAM, THIS FLOW WILL BE DIRECTED INTO THE SCRAM DISCHARGE VOLUME. FOLLOWING SCRAM, THIS FLOW WILL DECLINE AS VALVE F002 CLOSES AND AS THE SCRAM DISCHARGE VOLUME PRESSURIZES TO EQUAL THE REACTOR PRESSURE. AFTER THE SCRAM DISCHARGE VOLUME AND THE REACTOR VESSEL PRESSURE HAVE EQUALIZED, FLOW WILL BE DIVERTED TO THE REACTOR VESSEL VIA THE CRD WITHDRAW LINES AT A FLOW RATE DEPENDENT ON THE REACTOR PRESSURE:
   I.E. (A.) APPROX. 15 GPM AT "0" PSIG. REACTOR PRESSURE.
   (B.) APPROX. 6 GPM AT "1000" PSIG. REACTOR PRESSURE.
10. THIS VALUE APPLIES IMMEDIATELY FOLLOWING COMPLETION OF SCRAM. PRESSURE WILL SUBSEQUENTLY EQUALIZE WITH REACTOR PRESSURE.
11. DESIGN PRESSURE AND TEMPERATURE SHOWN IN "TABLE I" IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.
12. ALL VALUES SHOWN IN MODES A, B, C, AND D ARE NOMINAL UNLESS-OTHERWISE NOTED.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE SYSTEM (PROCESS DIAGRAM)	FIGURE 4.6-6c

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Description

The nuclear pressure relief system consists of safety/relief valves located on the main steam lines between the reactor vessel and the first isolation valve within the drywell.

These valves protect against overpressure of the nuclear system.

The safety/relief valves provide three main protection functions:

a. Overpressure relief operation. The valves open automatically to limit a pressure rise.

b. Overpressure safety operation. The valves function as safety valves and open (self-actuated operation if not already automatically opened for relief operation) to prevent nuclear system overpressurization.

c. Depressurization operation. The ADS valves open automatically as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier. The location and number of the ADS valves can be determined from Figure 5.2-6.

Chapter 15 discusses the events which are expected to activate the primary system safety/relief valves. The Chapter also summarizes the number of valves expected to operate during the initial blowdown of the valves and the expected duration of this first blowdown. For several of the events it is expected that the lowest set safety/relief valve will reopen and reclose as generated heat drops into the decay heat characteristics. The pressure increase and relief cycle will continue with lower frequency and shorter relief discharges as the decay heat drops off and until such time as the RHR system can dissipate this heat. The duration of each relief discharge should, in most cases, be less than 30 seconds. Remote manual actuation of the valves from the control room is recommended to minimize the total number of these discharges, with the intent of achieving extended valve seat life.

A schematic of the main safety/relief valve is shown in Figure 5.2-10. It is opened by either of two modes of operation:

a. The spring mode of operation which consists of direct action of the steam pressure against a spring-loaded disk that will pop open when the valve inlet pressure force exceeds the spring force. Figure 5.2-9 diagrams the valve lift vs. time characteristic.

b. The power actuated mode of operation which consists of using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the spring force, even with valve inlet pressure equal to zero psig.

The pneumatic operator is so arranged that if it malfunctions it will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressure.

For overpressure safety/relief valve operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at setpoints designated in Table 5.2-2. In accordance with the ASME Code, full lift in this mode of operation is attained at a pressure no greater than 3% above the setpoint.

To prevent backpressure from affecting the spring lift setpoint, each valve is provided with a device to counteract the effects of backpressure which results in the discharge line when the valve is open and discharging steam.

The safety function of the safety/relief valve is a backup to the relief function described below. The spring-loaded valves are designed and constructed in accordance with ASME III, NB 7640 as safety valves with auxiliary actuating devices.

For overpressure relief valve operation (power actuated mode), each valve is provided with a pressure sensing device which operates at the setpoints designated in Table 5.2-2. When the set pressure is reached, it operates a solenoid air valve which in turn actuates the pneumatic piston/cylinder and linkage assembly to open the valve.

c. Drywell Pressure Measurement

The drywell is at a slightly positive pressure during reactor operation. The pressure fluctuates slightly as a result of barometric pressure changes and outleakage. A pressure rise above the normally indicated values will indicate the presence of a leak within the drywell.

d. Drywell Temperature Measurement

The drywell cooling system circulates the drywell atmosphere through heat exchangers (air coolers) to maintain the drywell at its designed operating temperature and also provides cooling water to the air coolers. An increase in drywell atmosphere temperature would increase the temperature rise in the service water passing through the coils of the air coolers. Thus, an increase in the service water temperature difference between inlet and outlet to the air coolers will indicate the presence of reactor coolant or steam leakage. Also, a drywell ambient temperature rise will indicate the presence of reactor coolant or steam leakage. A temperature rise in the drywell is detected by monitoring the drywell temperature at various elevations, inlet and outlet air to the coolers, and the closed cooling water temperature increase between inlet and outlet to the coolers.

e. Drywell Air Sampling

The drywell air sampling system is used to supplement the temperature, pressure, and flow variation method described previously to detect leaks in the nuclear system process barrier. The system continuously monitors the drywell atmosphere for airborne radioactivity. The sample is drawn from the drywell. A sudden increase of activity, which may be attributed to steam or reactor water leakage, is annunciated in the control room. (Refer to Containment Indications, 7.5.1.5).

Table 5.2-9 summarizes the actions taken by each leakage detection function. The table shows that those systems which detect gross leakage initiate immediate automatic isolation. The systems which are capable of detecting small leaks initiate an alarm in the control room. The operator can manually isolate the violated system or take other appropriate action.

f. Reactor Vessel Head Closure

The reactor vessel head closure is provided with double seals with a leak off connection between seals that is piped to the equipment drain sump. Leakage through the first seal is annunciated in the control room. When pressure between the seals increases, an alarm in the control room is actuated. The second seal then operates to contain the vessel pressure.

g. Reactor Water Recirculation Pump Seal

Reactor water recirculation pump seal leaks are detected by monitoring the drain line. Leakage, indicated by high flow rate, alarms in the control room. Leakage is piped to the equipment drain tank.

h. Safety/Relief Valves

Temperature sensors connected to a multipoint recorder are provided to detect safety/relief valve leakage during reactor operation. Safety/relief valve temperature elements are mounted, using a thermowell, in the safety/relief valve discharge piping several feet from the valve body. Temperature rise above ambient is annunciated in the main control room. See the nuclear boiler system piping and instrumentation diagram, Figure 5.1-3a, b, c.

i. Valve Packing Leakage

Valve stem packing leaks of power-operated valves in the nuclear boiler system, reactor water cleanup system, high pressure core spray, low pressure core spray, reactor core isolation cooling system, residual heat removal system, and recirculation system are detected by monitoring packing leakoff for high temperature and are annunciated by an alarm in the control room.

5.2.5.3 Indication in Control Room

Leak detection methods are discussed in 5.2.5.1. Details of the leakage detection system indications are included in 7.6.1.3.

5.2.5.4 Limits for Reactor Coolant Leakage

5.2.5.4.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain and equipment drain sumps. The criterion for establishing the total leakage rate limit is based on the makeup capability of the RCIC system. The total leakage rate limit is established at 30 gpm, 25 gpm identified and 5 gpm unidentified.

The total leakage rate limit is also set low enough to prevent overflow of the drywell sumps. The equipment sump and the floor drain sump collect all leakage. The equipment sump is drained by one 50 gpm pump and the floor drain sump is drained by two 50 gpm pumps.

5.2.5.4.2 Normally Expected Leakage Rate

The pump packing glands, valve stems, and other seals in systems that are part of the reactor coolant pressure boundary and from which normal design leakage is expected are provided with drains or auxiliary sealing systems. Nuclear system valves and pumps inside the drywell are equipped with double seals. Leakage from the primary recirculation pump seals is piped to the equipment drain sump as shown in Figure 5.4-2b. Leakage in the discharge lines from the main steam safety/relief valves is monitored by temperature sensors that transmit a signal to the control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage.

Thus, the leakage rates from pumps, valve seals, and the reactor vessel head seal are measurable during plant operation. These leakage rates, plus any other leakage rates measured while the drywell is open, are defined as identified leakage rates.

5.2.5.5 Unidentified Leakage Inside the Drywell

5.2.5.5.1 Unidentified Leakage Rate

The unidentified leakage rate is the portion of the total leakage rate received in the drywell sumps that is not identified as previously described. A threat of significant compromise to the nuclear system process barrier exists if the barrier contains a crack that is large enough to propagate rapidly (critical crack length). The unidentified leakage rate limit must be low because of the possibility that most of the unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

An allowance for leakage that does not compromise barrier integrity and is not identifiable is made for normal plant operation.

The unidentified leakage rate limit is established at 5 gpm rate to allow time for corrective action before the process barrier could be significantly compromised. This 5 gpm unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Figure 5.2-13). Safety limits and safety limit settings are discussed in Chapter 16.

5.2.5.5.2 Sensitivity and Response Times

Sensitivity, including sensitivity tests and response time of the leak detection system are covered in Table 7.6-7.

5.2.5.5.3 Length of Through-Wall Flaw

Experiments conducted by GE and Battelle Memorial Institute, (BMI), permit an analysis of critical crack size and crack opening displacement (5.2.6 Ref. 5.2-4). This analysis relates to axially oriented through-wall cracks.

a. Critical Crack Length

Satisfactory empirical expressions have been developed to fit test results. A simple equation which fits the data in the range of normal design stresses (for carbon steel pipe) is:

The results given are for a longitudinally oriented flaw at normal operating hoop stress. A circumferentially oriented flaw could be subjected to stress as high as the 550°F yield stress, assuming high thermal expansion stresses exist. It is assumed that the longitudinal crack, subject to a stress as high as 30,000 psi, constitutes a "worst case" with regard to leak rate versus critical size relationships. Given the same stress level, differences between the circumferential and longitudinal orientations are not expected to be significant in this comparison

Figure 5.2-13 shows general relationships between crack length, leak rate, stress, and line size, using the mathematical model described previously. The asterisks denote conditions at which the crack opening displacement if 0.1 inch, at which time instability is imminent as noted previously under "Leakage Flow Rate". This provides a realistic estimate of the leak rate to be expected from a crack of critical size. In every case, the leak rate from a crack of critical size is significantly greater than the 5 gpm criterion.

If either the total or unidentified leak rate limits are exceeded, an orderly shutdown would be initiated and the reactor would be placed in a cold shutdown condition within 24 hours.

5.2.5.5.4 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in 5.2.5.5.3. Figure 5.2-13 shows general relationships between crack length, leak rate, stress and line size using the mathematical model.

5.2.5.5.5 Criteria to Evaluate the Adequacy and Margin of the Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier located in the drywell, reactor building and auxiliary building as shown in Table 5.2-9. The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system, if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened with significant compromise.

The leak detection system will satisfactorily detect unidentified leakage of 5 gpm.

Sensitivity, including sensitivity testing and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded, are covered in Table 7.6-7.

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

Section 5.2.5.1 describes the systems that are monitored by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in 5.2.5.1 and 7.6.1.3.

5.2.5.7 Sensitivity and Operability Tests

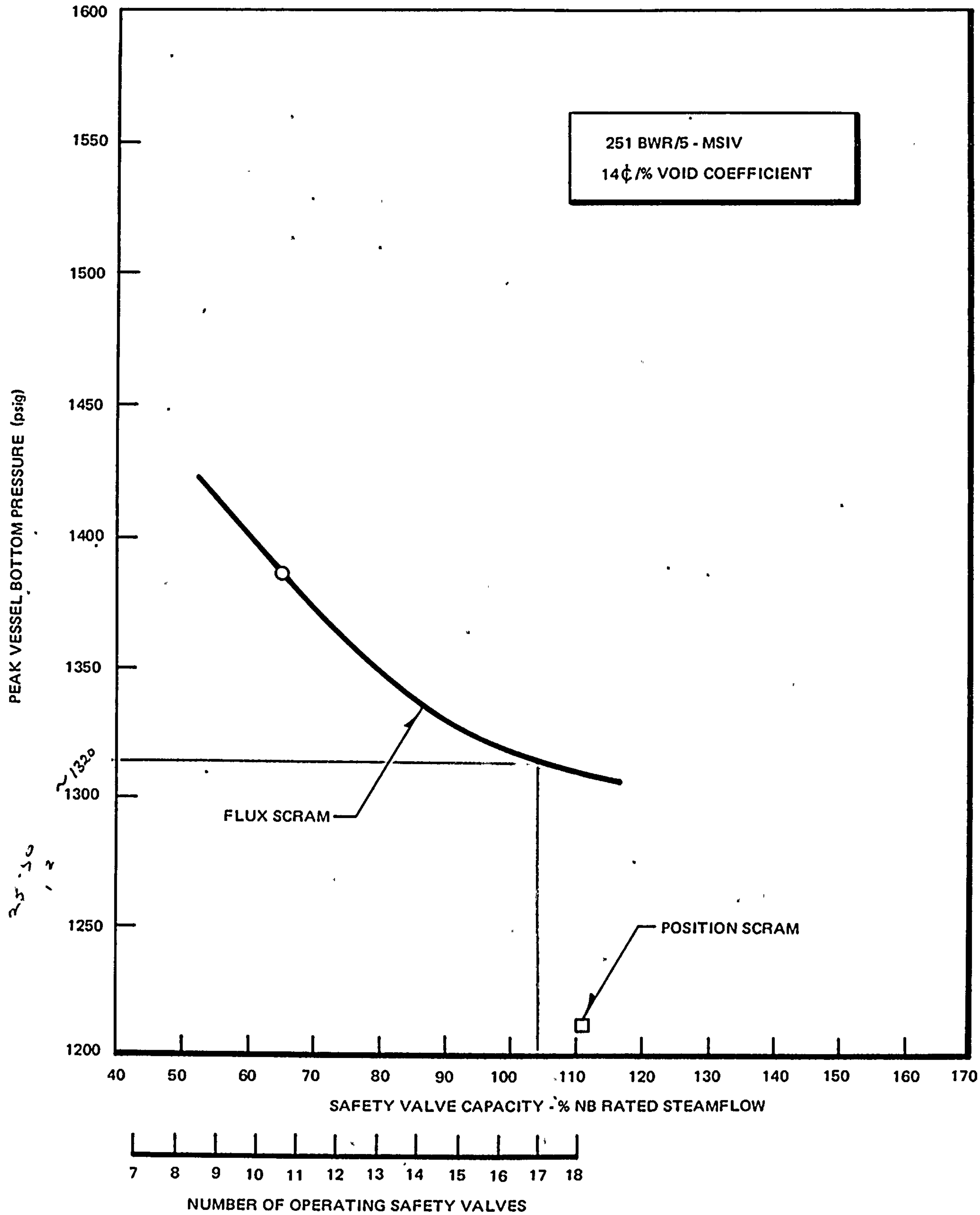
Testability of the leakage detection system is contained in 7.6.

5.2.5.8 Safety Interfaces

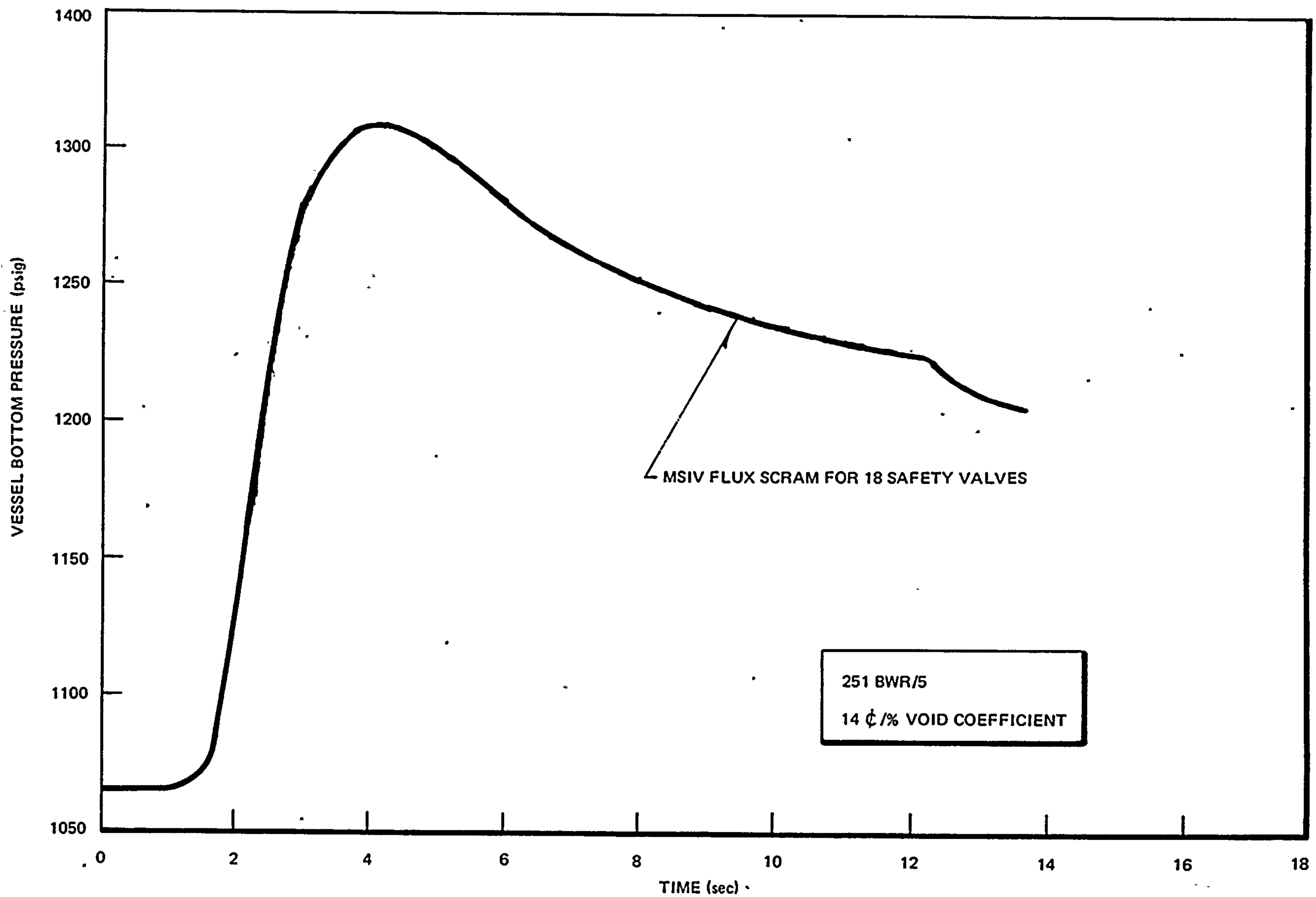
The Balance of Plant-GE Nuclear Steam Supply System safety interfaces for the leak detection system are the signals from the monitored balance of plant equipment and systems which are part of the nuclear system process barrier, and associated wiring and cable lying outside the Nuclear Steam Supply Equipment. These balance-of-plant systems and equipment include the main steam line tunnel, the safety/relief valves, and the turbine building sumps.

5.2.5.9 Testing and Calibration

Provisions for Testing and Calibration of the leak detection system are covered in Chapter 14, "Initial Tests and Operation".



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PEAK VESSEL PRESSURE VERSUS SAFETY/RELIEF CAPACITY	FIGURE 5.2-4



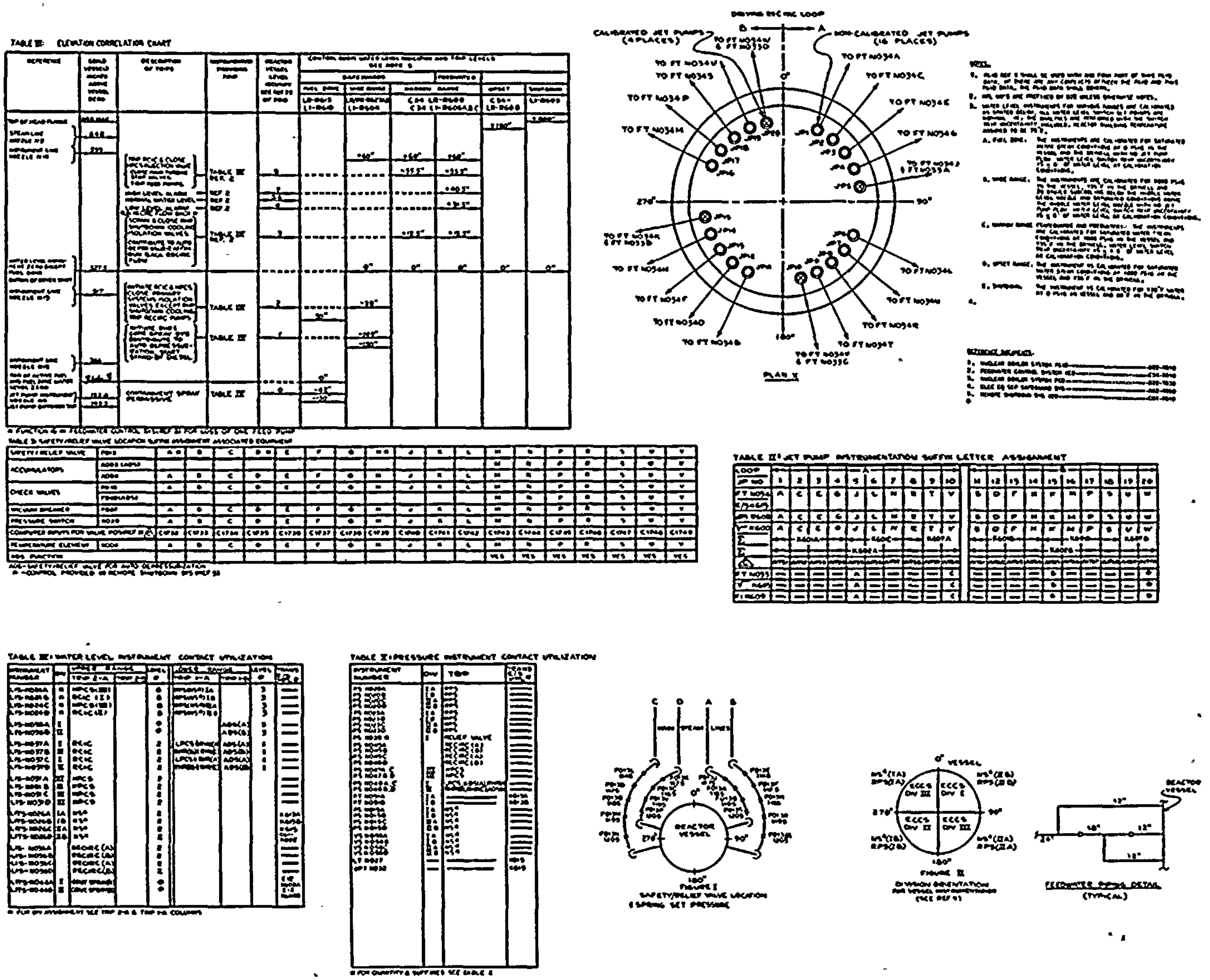
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

PEAK VESSEL PRESSURE VERSUS TIME FOR SAFETY VALVE CAPACITY SIZING TRANSIENT

FIGURE 5.2-5

AMENDMENT NO. 11
September 1980

AMENDMENT NO. 20
November 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM (P&ID)	FIGURE 5.2-6

5.4.6.2.1.3 Interlocks

The following defines the various electrical interlocks:

a. There are four key locked valves namely F063, F008, F068 and F069 and two key locked resets namely the "Isolation Resets".

b. F031's limit switch activates when full open and closes F010, F022, and F059.

c. F068's limit switch activates when full open and clears F045 permissive so F045 can open.

d. F045's limit switch activates when F045 is not fully closed and energizes 15 sec time delay for low pump suction pressure trip and also initiates startup ramp function. This ramp resets each time F045 is closed.

e. F045's limit switch activates when fully closed and permits F004, F005, F025, and F026 to open and closes F013 and F019.

f. The turbine trip throttle valve (part of C002) limit switch activates when fully closed and closes F013 and F019.

g. The combined pressure switches at reactor low pressure and high drywell pressure when activated closes F080 and F086.

h. 120% overspeed, high turbine exhaust pressure, low pump suction pressure, the combined signals of F045 fully open plus reactor high water level, or an isolation signal actuates and closes the turbine trip throttle valve. When signal is cleared, the trip throttle valve must be reset from control room.

i. 125% overspeed trips both the mechanical trip at the turbine and the trip throttle valve. The former is reset at the turbine and then the latter is reset in the control room.

j. An isolation signal closes F008, F063, F064, F076 and other valves as noted above in items f and h.

k. An initiation signal opens F010 if closed, F013 and F045; starts barometric condenser vacuum pump; and closes F022 and F059 if open.

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k. An initiation signal opens F010 if closed, F013 and F045; starts barometric condenser vacuum pump; and closes F022 and F059 if open.

l. High and low inlet RCIC steam line drain pot levels, respectively, open and close F054.

m. The combined signal of low flow plus discharge pressure open and with increased flow closes F019. Also see items e and f above.

n. The signal of in-line reserve tank low water level opens valve F031.

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions

Operating parameters for the components of the RCIC Systems, defined below, are shown on Figure 5.4-10.

a. One 100% capacity turbine and accessories

b. One 100% capacity pump assembly and accessories

c. Piping, valves, and instrumentation for:

    1. Steam supply to the turbine

    2. Steam supply to RHR condensing heat exchanger

    3. Turbine exhaust to the suppression pool

    4. Make-up supply from the condensate storage tank to the pump suction

    5. Make-up supply from the suppression pool to the pump suction

    6. Make-up supply from the RHR steam condensing heat exchangers

    7. Pump discharge to the head cooling spray nozzle, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

The basis for the design conditions was the American Society of Mechanical Engineers (ASME) Section III, Nuclear Power Plant Components.

l. High and low inlet RCIC steam line drain pot levels, respectively, open and close F054.

m. The combined signal of low flow plus discharge pressure open and with increased flow closes F019. Also see items e and f above.

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   5. Make-up supply from the suppression pool to the pump suction

   6. Make-up supply from the RHR steam condensing heat exchangers

   7. Pump discharge to the head cooling spray nozzle, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

The basis for the design conditions was the American Society of Mechanical Engineers (ASME) Section III, Nuclear Power Plant Components.

5.4.6.2.2.2 Design Parameters

Design parameter for the RCIC system components are listed below. See Figure 5.4-9 for cross-reference of component numbers listed below:

a. RCIC Pump Operation (C001) — (Refer to Figures 5.4-19a and 5.4-19b)

Flow rate	Injection Flow - 600 gpm Cooling Water Flow - 16-25 gpm Total Pump Discharge - 625 gpm (Includes no margin for Pump Wear)
Water Temperature Range	40°F to 140°F
NPSH	21 ft minimum
Developed Head	2890 ft @ 1155 psia Reactor Pressure 610 ft @ 165 psia Reactor Pressure
BHP, Not to Exceed	725HP @ 2890 feet Developed Head 130HP @ 610 feet Developed Head
Design Pressure	1500 psia
Design Temperature	40°F to 140°F

b. RCIC Turbine Operation (C002)

	H.P. Condition	L.P. Condition
Reactor Press (Sat. Temp.)	1155 psia	165 psia
Steam Inlet Pressure	1140 psia	150 psia
Turbine Exhaust Press	15 to 25 psia	15 to 25 psia
Design Inlet Pressure	1250 psia + saturated temperature	
Design Exhaust Pressure	165 psia + saturated temperature	

Steam Supply Isolation Valves (F008)	Open and/or close against full differential pressure of 1140 psi at a minimum rate of 12 inches per minute.
RHR Steam Supply Isolation Valves (F063/F064)	Open and/or close against full differential pressure of 1140 psi.
Cooling Water Pressure Control Valve (F015)	Self-contained downstream sensing control valve capable of maintaining constant downstream pressure of 75 psia.
Pump Suction Relief Valve (F017)	100 psig relief setting; 10 gpm at 10 percent accumulation.
Cooling Water Relief Valve (F018)	Sized to prevent over pressurization piping, valves a equipment in the coolant loop in the event of failure of pressure control valve F015.
Pump Test Return Valve (F022)	Is capable of throttling against 1000 psi differential pressure.
Relief Valve Barometric Condenser (F033)	Relief valve is capable of retaining 10 inches of mercury vacuum at 140°F ambient, with a set pressure of 5-7 psig and a flow of 20 gpm at 10 percent accumulation.
Pump Suction Valve Suppression Pool (F031)	Is located as close as practical to the primary containment.
Pump Suction Valve Condensate Storage Tank (F010)	Open and/or close against full differential pressure of 45 psi within 15 seconds.

Testable Check Valve (F065/F066)	System test mode bypasses this valve, and its functional capability shall be demonstrated separately. Therefore, valve test provisions are provided, including limit switches to indicate disc movement. The valve and valve associated equipment are capable of proper functional operation during high ambient conditions.
Warm-Up Line Isolation Valve (F076)	Valve will open and/or close against full differential pressure of 1140 psi at a minimum rate of 12 inches per minute.
Vacuum Breaker Valves (F080 & F086)	Valves will open and/or close against a differential pressure of 200 psi at a minimum rate of 12 inches per minute.

e. Rupture Disc

Assemblies (D001/D002)	Utilized for turbine casing protection, shall include a mated vacuum support to prevent rupture disc reversing under vacuum conditions. Rupture pressure 150 psig $\pm$ 10 psig Flow capacity 60,000 lb/Rv @ 165 psig

f. Condensate Storage

Requirements	Total reserve storage for RPV makeup is 135,000 gallons.

p. Follow steps n through s of 5.4.6.2.5.1.

5.4.6.2.5.3 Steam Condensing (Hot Standby) Operation

This mode of operation is manually initiated by the operator as follows:

a. Verification made in steps a through j of 5.4.6.2.5.1 shall be completed.

b. When the reactor water level is going to be maintained in the hot standby mode and the level starts to drop the RCIC system can be started by manually pushing the RCIC "Manual Initiation" push button. See 5.4.6.2.5.1(k) for RCIC subsequent starts. Concurrently, the RHR System water quality should be readied for vessel injection, see 5.4.7.2.6(b).

c. Adjust controller so it may be switched to manual mode and maintain same flow at pressure condition established by step b above. Then switch to manual mode.

d. Adjust flow controller set point as required to maintain desired reactor water level.

e. When RHR water is ready for vessel injection open RHR suction valve to RCIC System pump. During steam condensing operation if the RHR produces more condensate than required to maintain reactor level, the excess may be dumped to the suppression pool via the RHR system. Also, if more flow is required than supplied from the RHR head exchangers it will come from the condensate storage tank.

f. When steam condensing is completed and the RCIC system is no longer required, close the RHR suction valve, manually trip the RCIC system, and turn flow controller back to automatic.

g. Follow steps n through s of 5.4.6.2.5.1.

5.4.6.2.5.4 Limiting Single Failure

The most limiting single failure in the combined function of RCIC and HPCS systems is the failure of HPCS. If the capacity of RCIC System is adequate to maintain reactor water level, the operator follows 5.4.6.2.5.1. If however, the RCIC capacity is inadequate 5.4.6.2.5.1 applies, but additionally the operator may also initiate the ADS system described in 6.3.2.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions used in evaluating the RCIC system are presented in Chapter 15, "Accident Analyses," and Appendix A to Chapter 15, "Plant Nuclear Safety Operational Analyses." The RCIC system provides the flows required by the analysis (see Figure 5.4-10) within a 30 second interval based upon considerations noted in 5.4.6.2.4.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14, "Initial Test Program."

5.4.6.4 Safety Interfaces

The Balance-of-Plant/GE Nuclear Steam Supply System safety interfaces for the Reactor Core Isolation Cooling System are: (1) preferred water supply from the condensate storage tanks; (2) all associated wire, cable, piping, sensors, and valves which lie outside the Nuclear Steam Supply System scope of supply; and (3) air supply for testable check and solenoid actuated valve(s).

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of three independent loops. Each loop contains its own motor-driven pump, piping, valves, instrumentation and controls. Each loop has a suction source from the suppression pool and is capable of discharging water to the reactor vessel via a separate nozzle, or back to the suppression pool via a full flow test line. In addition, the A and B loops have heat exchangers which are cooled by standby service water. Loops A and B can also take suction from the reactor recirculation system suction, and can discharge into the reactor recirculation discharge or to the suppression pool and drywell spray spargers. Spool piece interties are provided to permit the RHR heat exchangers to be used to supplement the cooling capacity of the fuel pool cooling system. Spool piece interties are provided to permit the RHR heat exchangers to be used to supplement the cooling system. The A and B loops also have connections to reactor steam via the RCIC steam line and can discharge condensate to the RCIC pump suction or to the suppression pool. LaSalle 1 and 2, and Zimmer 1 are nuclear plants which employ similar RHR systems and which are in the process of being licensed.

5.4.7.1.1 Functional Design Basis

The RHR system has five subsystems, each of which has its own functional requirements. Each subsystem will be discussed

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6.2.1.1.5.2 Reactor Blowdown Conditions and Operator Response

In the highly unlikely event of a primary system leak in the drywell accompanied by a simultaneous open bypass path between the drywell and suppression chamber, several postulated conditions may occur. For a given primary system break area, the maximum allowable leakage capacity can be determined when the containment pressure reaches the design pressure at the end of reactor blowdown. The most limiting conditions would occur for those primary system break sizes which do not cause rapid reactor depressurization, but rather have long leakage duration. This break size, less than approximately 0.4 square feet, requires operator action to terminate the reactor blowdown.

Immediately after the postulated conditions given for a small primary system break, there would be a fairly rapid rise in containment pressure as the noncondensable gases in the drywell are carried over to the suppression chamber. During this portion of the transient, it is assumed that the plant operators are unaware that a leakage path exists. Under these circumstances, the maximum pressure that can occur in the suppression chamber is approximately 27 psig. This is the pressure that would result if all of the noncondensable gases initially in the containment are carried over to the suppression chamber free space. For the maximum allowable leakage calculations, it was assumed that the plant operators realize a leakage path exists only when the suppression chamber pressure reaches 30 psig. For conservatism, an additional ten minute delay is assumed before any corrective action is taken to terminate the transient. The corrective action is also assumed to take five minutes to be effective. At that time, the containment pressure would be equal to the design pressure if the allowable leakage had occurred. The specific corrective action taken after ten minutes is not accounted for in the analysis. The operators have several options available to them. If the source of the leakage is undefined, they would probably depressurize the primary system via either the main condenser or relief valves, or they could activate the containment spray system.

6.2.1.1.5.3 Analytical Assumptions

When calculating the allowable leakage capacities for a spectrum of break sizes, the following assumptions are made:

a. Flow through the postulated leakage path is pure steam. For a given leakage path, if the leakage flow consists of a mixture of liquid and vapor, the total leakage mass flowrate is higher but the steam flowrate is less than for the case of pure steam leakage. Since only the steam entering the suppression chamber free space results in the additional containment pressurization, this is a conservative assumption.

b. There is no condensation of the leakage flow on either the suppression pool surface or the containment and vent system structures. Since condensation acts to reduce the suppression chamber pressure, this is a conservative assumption. For an actual containment there will be condensation, especially for the larger primary system break where vigorous agitation at the pool surface will occur during blowdown.

6.2.1.1.5.4 Analytical Results

The containment has been analyzed to determine the allowable leakage between the drywell and suppression chamber. Figure 6.2-17a shows the allowable leakage capacity ($A/\sqrt{K}$) as a function of primary system break area. A is the area of the leakage flow path and K is the total geometric loss coefficient associated with the leakage flow path.

Figure 6.2-17a is a composite of two curves. If the break area is greater than approximately 0.4 square feet, natural reactor depressurization will rapidly terminate the transient. For break areas less than 0.4 square feet, however, continued reactor blowdown limits the allowable leakage to small values. The maximum allowable leakage capacity is at $A/\sqrt{K}$ = 0.026 square feet. Since a typical geometric loss factor would be three or greater, the maximum allowable flow path would be about 0.052 square feet. This corresponds to a 3 inch line size.

Burns and Roe, Inc. confirmed the results of the above analysis by GE in Reference 6.2-7. Further investigation into the transient nature of the problem was then undertaken at the request of the NRC.

A transient analysis using the CONTEMPT-LT (Ref. 6.2-8) computer code was performed. The code was modified to include the mass and energy transfer to the suppression pool from relief valve discharge. The limiting case was a very small reactor system break which would not automatically result in reactor depressurization. For this limiting case, it was assumed that the response of the plant operators was to shut the reactor down in an orderly manner at 100°F/hr cooldown rate. No other operator action was accounted for. Heat sinks considered were such items as major support steel inside containment, the reactor pedestal, the diaphragm floor and support columns and the steel and concrete of the primary containment. Based on this analysis, the allowable bypass leakage ($A/\sqrt{K}$) was 0.028 ft^2. The drywell pressure transient is shown in Figure 6.2-17b along with the corresponding curves of wetwell pressure, wetwell temperature and suppression pool temperature.

The allowable bypass leakage of 0.028 ft^2 is well above the maximum possible containment bypass leakage. Periodic testing will be performed to confirm that the containment bypass leakage does not exceed $A/\sqrt{K}$ = 0.0045 ft^2. Figure 6.2-17c presents the resulting containment transient for $A/\sqrt{K}$ = 0.0045 ft^2. The peak containment pressure shown in Figure 6.2-17c is well below the containment design pressure.

6.2.1.1.6 Suppression Pool Dynamic Loads

A generic discussion of the suppression pool dynamic loads and asymmetric loading conditions is given in Mark II Dynamic Forcing Function Information Report, Reference 6.2-4. A unique plant assessment of these dynamic loads is made in WNP-2 Design Assessment Report, Reference 6.2-5.

6.2.1.1.7 Asymmetric Loading Conditions

See comment in 6.2.1.1.6.

All containment purge valves, including the 2" bypass valves, are designed to shut within four seconds of receipt of a containment isolation signal and to shut against full containment design pressure, 45 psig. The containment isolation signals and the purge valves are part of the containment isolation system which is an ESF system. Each purge line has two isolation valves. These valves are opened by allowing compressed air to oppose a spring in the valve actuator. On a loss of compressed air, loss of electrical signal, or on a containment isolation signal the valve is shut. If the purge system were operating at the time of a LOCA, the system will automatically be secured. The level of the activity released through the purge system before isolation would be limited to the activity present in the coolant prior to the accident since the purge system will be isolated before any postulated fuel failure could occur.

6.2.1.1.8.3 Post - LOCA

The unit coolers are not required after a LOCA since heat removal is then accomplished by the containment cooling system, a subsystem of the RHR system, as described in 6.2.2. Two 100% redundant hydrogen recombiners are available to be placed in operation to ensure that the hydrogen buildup does not reach a flammable level. Containment purge has the capability for a controlled purge of the containment atmosphere to aid in hydrogen control, if necessary.

Any equipment located inside the primary containment which is required to operate subsequent to a LOCA has been designed to operate in the worst anticipated accident environment for the required period of time.

6.2.1.1.9 Post Accident Monitoring

A description of the post accident monitoring systems is provided in 7.5.

6.2.1.2 Containment Subcompartments

The two areas within the primary containment considered subcompartments are the area within the sacrificial shield wall and the area above the refueling bulkhead plate at elevation 583'.

All potential pipe breaks within the sacrificial shield wall have been evaluated. The information is contained in References 3.8-5, 3.8-6, 3.8-7, and 3.8-24. These references have been previously submitted to the NRC.

Two analyses were performed to ensure the adequacy of the refueling bulkhead and inner refueling bellows at elevation 583'. The first analysis, a break of the RCIC head spray line, determines the maximum downward loading due to pipe breaks, and the second analysis, a break of the RRC suction line, determines the maximum upward loading. These analyses are summarized below.

Subcompartment analyses for a postulated high energy pipe break in the primary containment were performed for the annulus inside the sacrificial shield wall, and the regions above and below the bulkhead plate which divides the drywell into the upper head region and the lower region.

The analyses for the annulus were reported in full detail in References 6.2-9 through 6.2-11. The result of the case of a 60-node model of the shield wall annulus for pressure transient calculation was confirmed by the NRC, and the analysis was considered acceptable for the shield wall base design and the design of the shield wall above the base, as stated in NRC letters (References 6.2-12 and 6.2-13).

Information with respect to the analyses for the upper head region and the lower region is provided below.

a. For the subcompartment analysis in the upper head region, the worst case is a double ended guillotine break in the 6" RCIC line above the RPV head at approximately elevation 595.1 ft. For the analysis in the lower region in order to determine the differential pressure across the bulkhead plate, the worst case is a double ended guillotine break in the 24" recirculation line anywhere inside the drywell. The pipe breaks were postulated for the subcompartment structural design and the component support design.

b. The blowdown mass and energy release rates as functions of time for the 6" RCIC line break are shown in Table 6.2-20 (Steam) and 6.2-21 (Water). The blowdown mass and energy release rates as functions of time for the 24" recirculation line break are shown in Table 6.2-22 (Steam) and 6.2-23 (Water).

The design specifications require each isolation valve to be operable under the most severe operating conditions that it might experience. Each isolation valve is afforded protection by separation and/or adequate barriers from the consequences of potential missiles.

Electrical redundancy is provided in isolation valve arrangements which eliminates dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line have been routed separately. Cables have been selected and based on the specific environment (see 3.11) to which they may be subjected, such as high radiation, high temperature, and high humidity.

Provisions for administrative control and/or locks ensure that the position of all nonpowered isolation valves is maintained and known. For all power operated valves the position is indicated in the main control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

6.2.4.3.2 Evaluation Against General Design Criteria

6.2.4.3.2.1 Evaluation Against Criterion 55

The reactor coolant pressure boundary (RCPB), as defined in 10 CFR 50, Section 50.2 (v), consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valve. The lines of the reactor coolant pressure boundary which penetrate the containment include provisions for isolation of the containment, thereby precluding any significant release of radioactivity. Similarly, for lines which do not penetrate the containment but which form a portion of the reactor coolant pressure boundary, the design ensures that isolation of the reactor coolant pressure boundary can be achieved.

6.2.4.3.2.1.1 Influent Lines

Influent lines which penetrate the primary containment and connect directly to the RCPB are equipped with at least two isolation valves, one inside the drywell, and the other as close to the external side of the containment as practical. Protection of the environment is provided by these isolation valves.

Table 6.2-16 contains those influent pipes that comprise the reactor coolant pressure boundary and penetrate the containment.

6.2.4.3.2.1.1.1 Feedwater Lines

The feedwater lines are part of the reactor coolant pressure boundary as they penetrate the drywell to connect with the reactor pressure vessel. The isolation valve inside the drywell is a y-pattern check valve, located as close as practicable to the containment wall. Outside the containment is another y-pattern check valve located as close as practicable to the containment wall and farther away from the containment is a motor operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. However, in case a loss-of-coolant accident occurs without a seismic event, the design allows the condensate and condensate booster pumps to supply feedwater to the vessel through a bypass line around the reactor feed pumps - which are tripped on a loss of steam supply - as soon as the vessel is partially depressurized. For this reason, the outermost gate valve does not automatically isolate upon signal from the protection system. The gate valve meets the same environmental and seismic qualifications as the outboard isolation valve. The valve is capable of being remotely closed from the control room to provide long-term leakage protection upon operator judgement that feedwater makeup is unavailable or unnecessary. No credit is taken for feedwater flow in accessing core and containment response to a loss-of-coolant accident.

6.2.4.3.2.1.1.2 HPCS Line

The HPCS line penetrates the drywell to inject directly into the reactor pressure vessel. Isolation is provided by an air testable check valve, located inside the drywell with position indicated in the main control room, and remote-manually actuated gate valve located as close as practicable to the exterior wall of the containment. Long-term leakage control is maintained by this gate valve. If a loss-of-coolant accident occurred, this gate valve would receive an automatic signal to open.

6.2.4.3.2.1.1.3 LPCI and LPCS Lines

Satisfaction of isolation criteria for the LPCI and the LPCS system is accomplished by use of remote-manually operated gate valves and check valves. Both types of valves are normally closed with the gate valves receiving an automatic

(see Table 3.2-1). The skid and the equipment mounted on it meet Seismic Category I requirements. The system is designed to be in accordance with IEEE Std. 279-1971, (Criteria for Protection Systems for Nuclear Power Generating Stations), and IEEE 344-1971 (Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generation Stations). The system is designed to withstand dynamic effects present in the containment (temperature and pressure) following the occurrence of a loss-of-coolant accident. All skid mounted components subjected to the containment gas stream are capable of withstanding the total post LOCA integrated radiation dose of 3.1×10^7 rads. The hydrogen recombiner system is used in conjunction with monitoring the atmosphere of the containment for hydrogen concentrations. The monitoring system is operated continuously. Readout is provided in the main control room.

Following the postulated LOCA, warmup of the hydrogen recombiner system is initiated from remote-manual controls. The system is sized to maintain the hydrogen concentration in the primary containment to less than 4% by volume. The system requires a 30 minute warmup period. The system is placed into operation manually from the main control room. Once placed into operation, the system continues to operate until manually shutdown after an adequate safety margin in hydrogen concentration is reached. The operation of the system is monitored from the main control room.

The containment atmospheric control system is supplied by redundant Class 1E power supplies. Cooling water systems are placed into operation by the same signals which start up the ECCS.

Cooling water for operation of the system (at 85°F) is taken from the standby service water system. This cooling water is used for the following purposes:

a. Scrubber (water consumption 1-10 gpm, average 4 gpm): removing particulate matter and condensing steam in the gases from the primary containment and reducing the temperature of these gases, and

b. Aftercooler (water consumption 20-50 gpm); cooling the gases leaving the recombiner prior to returning this mixture of gases and water vapor to the primary containment.

The cooling water supplied to the aftercooler is returned to the standby service water system. The cooling water supplied to the scrubber is discharged to the suppression pool.

All components of the containment atmosphere control system are redundant. Controls include the control panel located in the main control room and the local control panel for each recombiner located in environmentally suitable rooms in the reactor building. All of the functions necessary to control the system are located in the main control room.

6.2.5.2.4 Containment Purge

Containment purge, discussed in 6.2.1.1.8, has the capability for a controlled purge of the containment atmosphere to aid in hydrogen control, if necessary.

6.2.5.3 Design Evaluation

Based on the assumptions of the model described below, it is calculated that the hydrogen concentration in the drywell eventually reaches 4% by volume approximately 10.0 hours after the postulated LOCA if the hydrogen recombiner is not in operation. The recombiner is started, however, when the hydrogen concentration reaches approximately 3.5% by volume (2.75 hours after the postulated LOCA) to limit the hydrogen concentration below 4% by volume. Figure 6.2-26 shows the drywell and suppression chamber hydrogen concentration as a function of time, with and without operation of the hydrogen recombiner system at design capacity of 150 scfm and at 105 scfm, minimum flow required to maintain the hydrogen concentration below 4% by volume.

The determination of the time dependent hydrogen concentration in the drywell and suppression chamber atmospheres is based on a two-region model of the primary containment, a drywell and a suppression chamber atmosphere.

The drywell and suppression chamber free volumes contain air and water vapor at atmospheric pressure just prior to the postulated LOCA. Gases considered available for hydrogen dilution are the non-condensibles and water vapor present during normal operating conditions. Water vapor generated from blowdown is not considered. The radiolytic generation of free oxygen is added to the total inventory of gases. The pressure in containment is assumed to remain at atmospheric pressure and the temperature history of Figure 6.2-7 curve a, is used. The hydrogen contribution from zinc and organics took no credit for dilution.

6.2.6.5 Special Testing Requirements

The secondary containment shall be subjected to tests prior to initial fuel loading and at each refueling outage to assure the maximum allowable leakage rate of 100% of secondary containment free volume per day at -0.25 inches water gauge pressure with respect to outside atmospheric pressure. The test procedure for determining that the leakage rate does not exceed the maximum allowable is summarized in 6.2.3.4. See Chapter 16, Technical Specifications. Test procedures for the MSIV-LCS are in 6.7.

6.2.7 REFERENCES

6.2-1 James, A. J., "The General Electric Pressure Suppression Containment Analytical Model", April 1971, (NEDO-10320).

6.2-2 James, A. J., "The General Electric Pressure Suppression Containment Analytical Model", Supplement 1, May 1971 (NEDO-10320).

6.2-3 Moody, F. J., "Maximum Two-Phase Vessel Blowdown from Pipes", Topical Report APED-4824, General Electric Company, 1965.

6.2-4 "MK II Containment Dynamic Forcing Functions Information Report (Revision 2)", General Electric and Sargeant and Lundy, NEDO-21061, September 1976.

6.2-5 "Plant Design Assessment Report for SRV and LOCA Loads (Revision 2)", Washington Public Power Supply System, August 1979.

6.2-6 J. D. Duncan and J. E. Leonard, "Emergency Cooling in BWR's Under Simulated Loss-of-Coolant (BWR PLECMP) Final Report," FEAP-13197, General Electric, June 1971.

6.2-7 WPPSS Report, "Drywell to Wetwell Leakage Study", WPPSS-74-2-RS, July, 1974. (Submitted to NRC by WPPSS, Letter GO2-74-17, dated August 9, 1974).

6.2-8 Wheat, L. L.; Wagner, R. J.; Niederauer, G. F.; Obenchain, C. F., CONTEMPT-LT-- A COMPUTER PROGRAM FOR PREDICTING CONTAINMENT PRESSURE-TEMPERATURE RESPONSE TO A LOSS-OF-COOLANT ACCIDENT, ANCR-1219, Aeroject Nuclear Company, June, 1975.

6.2-9 Washington Public Power Supply System, Nuclear Project No. 2, Report No. WPPSS-74-2-R2-A, "Sacrificial Shield Wall Design Supplemental Information", February 11, 1975.

6.2-10 Washington Public Power Supply System, WPPSS Nuclear Project No. 2 Response to NRC comments, Report No. WPPSS-74-2-R2-A, "Sacrificial Shield Wall Design Supplemental Information", June 26, 1975.

TABLE 6.2-14

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TESTS

I ELECTRICAL PENETRATIONS

PENETRATION NUMBER	TYPE SERVICE	COMMENTS
X-100 A,B,C and D	Neutron Monitoring	Electrical Penetrations are provided with double seals and are separately testable at 45 psig. The test taps and seals are so located that tests can be conducted without entry into nor pressurization of the primary containment.
X-101 A,B,C and D	Control Rod Position Indicator	
X-102 A and B	Thermocouple and RTD	
X-103 A,B and C	Medium Voltage Power	
X-104 A,B,C and D	Low Voltage Power	
X-105 A,B,C and D	Control and Indication	
X-106 A,B,C and D	Spares	
X-107 A and B	Low Voltage Power Control and Indication.	

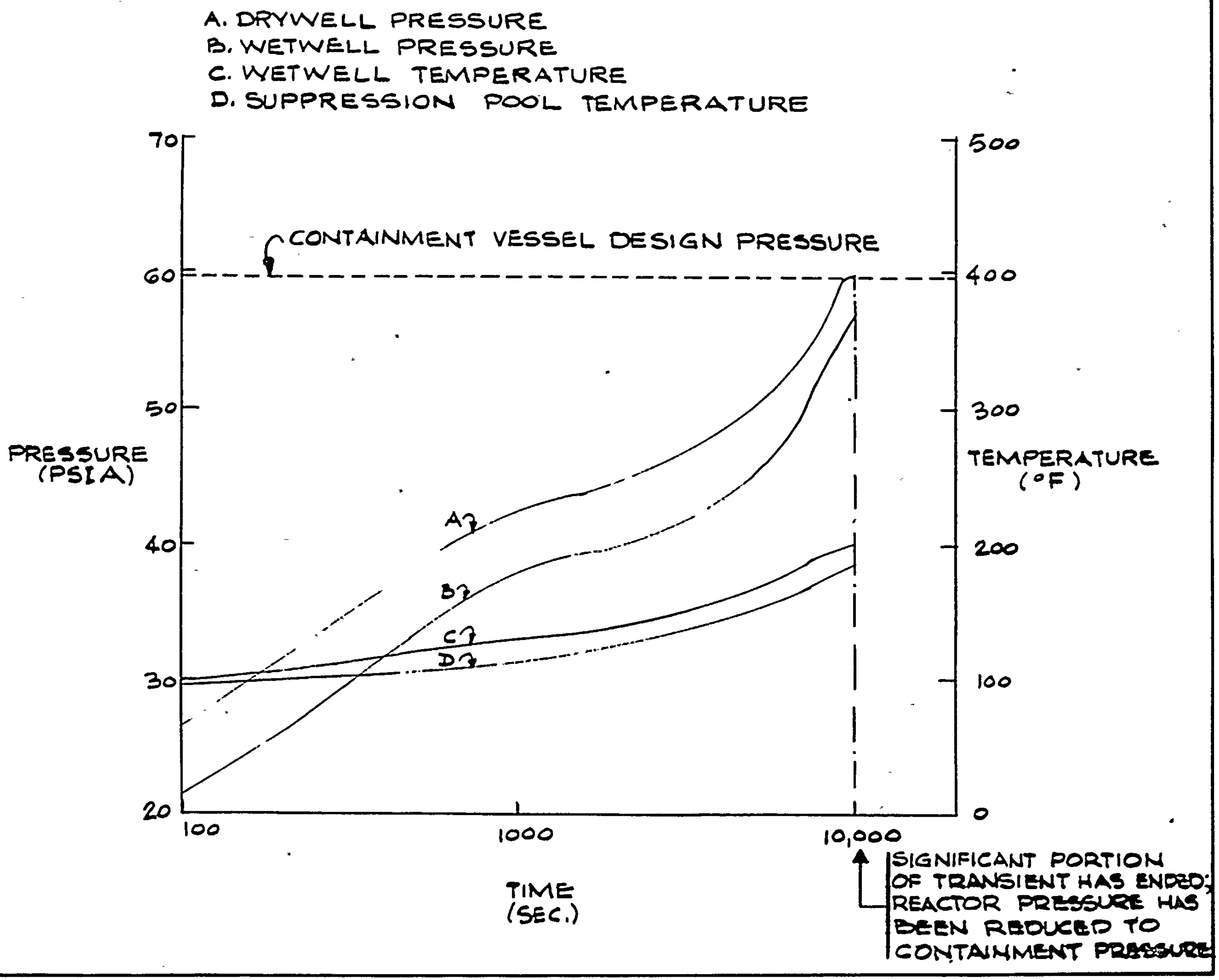
II PERSONNEL AND EQUIPMENT ACCESS PENETRATIONS

PENETRATION NUMBER	TYPE SERVICE	COMMENTS
X-15	Equipment Hatch	Separately testable at 45 psig without pressurization of the primary containment. Testing of the personnel access lock is described in detail in 3.8.2.7.5.
X-16	Personnel Access Lock	
X-28	CRD Removal Hatch	
X-51	Suppression Chamber Access Hatch	

TABLE 6.2-15

SUPPRESSION CHAMBER/REACTOR BLDG. DIFFERENTIAL PRESSURE INDICATING SWITCH CHARACTERISTICS

Instrument Tag No.	Range	Setpoint	Accuracy
CSP-dPIS-4	0-2 PSID	0.5 PSID	± 2%
CSP-dPIS-5	0-2 PSID	0.5 PSID	± 2%
CSP-dPIS-6	0-2 PSID	0.5 PSID	± 2%



Amendment No. 3
March 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTAINMENT TRANSIENT FOR MAXIMUM ALLOWABLE BYPASS CAPACITY, $A/\sqrt{K} = 0.028\ FT^2$

FIGURE 6.2-11b

6.4.3 SYSTEM OPERATIONAL PROCEDURES

As summarized in 6.4.2.2 and described in detail in 9.4.1, the response of the main control room habitability systems to any emergency condition is fully automatic. During normal and emergency operation the control room operator selects the air handling unit which operates to maintain design temperatures in the control room. Periodically the operating unit is stopped and the standby unit started so that the service time of both units is approximately equal. In the event the operating unit fails, control room personnel start the standby unit from the control room.

The responses of the control room habitability system to either chlorine or airborne radioactivity are compatible and fully automatic. On receipt of either signal the emergency filter units are started, the exhaust fan stopped and associated damper shut, the appropriate air intake isolation valves closed, and an alarm annunciated in the control room. Procedures are established for the donning of portable breathing apparatus by operating personnel on receipt of a high chlorine level alarm.

6.4.4 DESIGN EVALUATION

6.4.4.1 Radiological Protection

Operating personnel in the main control room are protected from the radiological effects of a loss-of-coolant accident by pressurizing the main control room with 1000 cfm of filtered air drawn from either of two remote fresh air intakes. This operation limits the 30 day dose to operators to below that of Criterion 19 of 10CFR50, Appendix A. All essential components of the control room habitability systems are redundant, Seismic Category I, and powered from Class 1E buses. The 30 day dose analysis for control room operators is summarized below.

The guidance in Revision 1 to USNRC Standard Review Plan (SRP) 6.4 was used in the control room dose analysis for WNP-2. The emergency ventilation system is of the dual inlet design with the automatic selection control feature. Since the design employs only two valves in series on each remote inlet, the damper repair alternative delineated in Appendix A to SRP 6.4, Revision 1, was utilized.

All components of the valves and operators for the remote inlets are considered to be highly reliable. The valve is a butterfly-type with the disc keyed to a pivot shaft. The

operator is an electro-hydraulic type with a spring driven failure mode to open. The valves are readily accessible to the control room operator through a ceiling hatch or via a stairwell to the floor above the control room. The manual manipulations necessary to reposition a failed closed valve have been investigated and found to be simple. For analysis purposes, a repair time of 1/2 hour is assumed. Simulated repairs and valve manipulations will be conducted as part of plant training. Valve position indication in the control room is positive for the operators via limit switches mounted directly to the valve shafts. In addition, each valve shaft is marked to indicate the full open and closed valve position at a glance so the operator has reliable local indication during manual repositioning. The assumptions made in the dose analysis are identical or more conservative than those made in 15.6.5, for the DBA-LOCA since this is the limiting event for control room impact. In summary,

a. the source term is comparable to Table 15.6-13;

b. primary to secondary containment leakage is assumed to be 0.73 weight %/day (0.5% per day is due to the design basis leak rate and 0.23%/day is due to the MSIV leakage control system);

c. the bypass leakage from the primary containment to the environment is assumed to be 0.74 scfh;

d. the standby gas treatment system in the reactor building is assumed to incur 7 scfm inleakage per unit bypassing the filter train. Though negligible inleakage is expected, and the integrity of the system is controlled by technical specification, 7 scfm is evaluated to give an upper bound on the control room dose consequences. The assumed inleakage results in an effective degradation of SB&T unit filter efficiency from 99% to 98.7%. See 6.5 for further discussion;

e. as per SRP 6.4, Revision 1, it is assumed a failure occurs in one of the two valves in series on the remote inlet furthest from the plant. No manual correction was assumed for two hours after the accident. Manual correction was assumed to require 1/2 hour and 1/2 hour was added for margin. Hence, both remote intakes are assumed to become operational three hours after the accident. During the first three hours, only the inlet closest to the reactor building vent is considered operational;

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LIST OF TABLES (Continued)

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8.3.1.4. Fire detection and protection in the areas where wiring is installed is described in 9.5.1.

i. Conformance to IEEE 387-1972 - Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations

Design and qualification testing of the standby power system used to furnish electrical power to safety loads conforms to IEEE 387 to ensure that system requirements for redundancy, single failure criteria, adequate capacity, capability and reliability are adequately met. The standby power source as an integrated system component satisfies the requirements of IEEE 308 as discussed in 8.3.

7.1.2.4 Conformance to Regulatory Guides

The following is a discussion of Regulatory Guides which apply equally to all safety-related systems described in Chapter 7. Those Regulatory Guides which do not apply equally to all safety-related systems are discussed for each system in the applicable analysis portion of 7.2, 7.3, 7.4, 7.5, and 7.6, and Appendix C.

a. Conformance to Regulatory Guide 1.11 (2/17/72)

All instrument lines penetrating or connected directly to the primary containment atmosphere, which are part of safety-related systems, meet the requirements of Regulatory Position C.1. This is accomplished by redundancy, independence, and by allowing for safety system testability, by line orificing or sizing and by including automatic line shutoff capability if line integrity is lost. Refer also to 6.2.4.3.1.

All other instrument lines that penetrate primary containment or are connected directly to the containment atmosphere meet Regulatory Position C.2.

b. Conformance to Regulatory Guide 1.29 (6/7/72)

All safety-related instrumentation and control equipment is classified as Seismic Category I, designed to withstand the effects of the safe shutdown earthquake (SSE) and remain functional during normal and accident conditions. Qualification and documentation procedures used for Seismic Category I equipment and systems are identified in 3.10 and Table 3.2-1.

c. Conformance to Regulatory Guide 1.30 (8/11/72)

The quality assurance requirements of IEEE 336-1971 (see discussion above) are applicable during the plant design and construction phases and will also be implemented as an operational QA program during plant operation in response to Regulatory Guide 1.30. The specific requirements of Regulatory Guide 1.30 are met as discussed in Chapter 17.

d. Conformance to Regulatory Guide 1.40 (3/16/73)

There are no safety-related continuous duty motors installed inside the primary containment.

e. Conformance to Regulatory Guide 1.47 (1973)

Each safety-related system described in 7.2, 7.3, 7.4, and 7.6 is provided with an automatically or operator initiated system level bypass and inoperability annunciator.

The system level annunciators are grouped together and located on a panel near the control room operator's console.

In addition to system level annunciation, component and channel level annunciators are provided on other panels either in the control room near system controls or locally near affected equipment, to indicate the cause of the system bypass or inoperability.

A switch is provided for manual actuation of each system level annunciator to allow display of those bypass or inoperable conditions which are not automatically indicated.

Typically, the following bypasses or inoperabilities cause actuation of system level (and component level) annunciation for the affected system:

1. Pump motor breaker not in operate position;
2. Loss of pump motor control power;
3. Loss of motor operated valve control power/motive power;
4. Logic power failure;
5. Logic in test;
6. System lineup improper;

9. Main Steam Line Radiation Monitors

High radiation in the vicinity of the main steam lines may indicate a gross fuel failure in the core. When high radiation is detected near the steam lines, a scram is initiated to limit the release of fission products from the fuel. This condition also signals the primary containment and reactor vessel isolation control system to initiate containment isolation to prevent the release of fission products. Refer to Figure 7.2-8 (Process Radiation Monitoring IED).

Main steam line radiation is monitored by rour redundant radiation monitors. Each monitor provides a trip signal to one of the four RPS sensor trip channels when high gamma radiation is detected in the vicinity of the main steam lines.

Main steam line high radiation occurs due to gross release of fission products from the fuel. This condition would only occur after an accident, in which the primary variables for trip initiation would be reactor vessel low water level, reactor vessel high pressure or power. These variables are diverse to main steam high radiation. In the event of a failure to scram from main steam high radiation, trip initiation, and protective action will have been initiated from these other diverse variables.

10. Manual Scram

A scram can be initiated manually. There are four scram switches, one for each of the four RPS trip logics. The manual scram switches are arranged in two groups of two switches. One group contains the A_1 and B_1 switches and A_2 and B_2 are in the other group. To initiate a manual scram, at least two switches in a group must be depressed. By operating the manual scram switch for one logic at a time and then resetting that logic, each actuator logic can be tested for manual scram capability.

11. Reactor Mode Switch Manual Scram

Even though the action is not a safety function, reactor scram can be initiated by placing the mode switch in the "shutdown" position. The mode switch consists of four independent banks of contacts. A "shutdown" position contact from each of the four banks is a scram input to the associated RPS trip logic. Refer to the RPS electrical schematics referenced in 1.7 for a complete description of the reactor mode switch.

The scram signal, initiated by placing the mode switch in SHUTDOWN is automatically bypassed after 10 seconds by a timer

which allows the control rod drive hydraulic system valve lineup to be restored to normal before the control room operator can reset the RPS logic.

The RPS receives power from two high inertia a-c motor generator sets. A flywheel provides high inertia sufficient to maintain voltage and frequency within 5% of rated values for at least 15 seconds following a total loss of power to the drive motor. The drive motor supplies are backed up by divisional diesel generator supplies which prevent automatic scram from a loss of offsite power.

Alternate power is available to each RPS bus and is manually switched to the bus as necessary for maintenance of the RPS M-G sets. The alternate power switch is interlocked to prevent simultaneous feeding of both buses from the same source. The switch also prevents paralleling of a motor-generator set with the alternate supply.

The RPS is designed to utilize a fail-safe logic and actuation scheme. Therefore, the power supplied by the RPS M-G sets to hold RPS components energized is expendable and considered non-safety-related. The M-G sets are not Quality Class I or Seismic Class I. However, to assure that overvoltage or underfrequency does not damage safety related components within the RPS, redundant Class IE bus monitoring devices are provided to trip the RPS bus should voltage and frequency exceed predetermined limits.

7.2.1.2 Design Basis

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are presented in that chapter.

The following variables are monitored in order to provide protective actions to the RPS indicating the need for reactor scram.

a. Variables Monitored to Provide Protective Actions.

   1. Neutron Monitoring System Trip

   2. Reactor Vessel System High Pressure

f. Range of Transient, Steady State, and Environmental Conditions

The reactor protection system (RPS) 120 Vac power is provided by high inertia M-G sets. Voltage regulation is designed to respond to a step load change of 50% of rated load with an output voltage change of not more than 15%. The flywheel on each M-G set provides stored energy to maintain voltage and frequency within ±5%, for 15 seconds, preventing momentary switchyard transients from causing a scram. RPS relays and contactors will operate without failure within the range of -15% to +10% of rated voltage. Refer also to the discussion in 7.2.1.1(3).

Environmental conditions for proper operation of the RPS components are discussed in Table 3.11-1 for inside the containment and Table 3.11-2 for outside the containment.

g. Malfunctions, Accidents, and Other Unusual Events Which Could Cause Damage to Safety Systems

Unusual events are defined as malfunctions, accidents, and others which could cause damage to safety systems. Chapter 15 and Appendix 15A, "Accident Analysis", describe the following credible accidents and events; floods, storms, tornados, earthquakes, fires, LOCA, pipe break outside containment, feedwater line break, and missiles. Each of these events is discussed below for the RPS.

All components essential to the operation of the RPS are designed, fabricated, and mounted to Class IE standards. However even though the sensors initiating reactor scram which monitor turbine stop valve position and turbine governor valve fast closure are designed and purchased to Quality Class I, Seismic Class I, they are physically mounted on equipment which is not Seismic Class I/Quality Class I, and located in the turbine generator building which is not Siesmic Class I.

For this reason other diverse variables (reactor pressure and neutron flux trips) may be relied upon for reactor scram if components in the turbine generator building fail.

1. Floods

The buildings containing RPS components have been designed to meet the PMF (Probable Maximum Flood) at the site location. This ensures that the buildings will remain water tight under PMF including wind generated wave action and wave runup. For a discussion of internal flooding protection refer to 3.4.1.4.1.2, 3.4.1.5.2, and 3.6.

2. Storms and Tornados

The buildings containing RPS components except the turbine generator building have been designed to withstand all credible meteorological events and tornados as described in 3.3.

3. Earthquakes

The structures containing RPS components except the turbine building have been seismically qualified as described in 3.7 and 3.8, and will remain functional during and following a safe shutdown earthquake (SSE).

4. Fires

To protect the RPS in the event of a postulated fire, the RPS trip logics have been divided into four separate sections within two separate RPS panels. The sections within a panel are separated by fire barriers. If a fire were to occur within one of the sections or in the area of one of the panels, the RPS functions would not be prevented by the fire. The use of separation and fire barriers ensures that, even though some portion of the system may be affected, the RPS will continue to provide the required protective action.

Within the control room PGCC (underfloor cable routing ducts) heat detectors and products of combustion detectors are provided to initiate a halon fire suppression system.

Throughout main plant areas redundant RPS cables are routed in separate wireways sufficiently separated from each other such that a fire cannot effect more than one RPS division.

5. LOCA

The following RPS system components are located inside the drywell and would be subjected to the effects of a design basis loss-of-coolant accident (LOCA):

a) Neutron monitoring system (NMS) cabling from the detectors to the main control room.

b) MSIV (inboard) position switches.

c) Reactor vessel pressure and reactor vessel water level instrument taps and sensing lines, which terminate outside the drywell.

To prevent inadvertent actuation of the ADS two channels of logic for each ADS trip system (A & B) are used. Both channels must be activated to actuate an ADS trip system. Refer to Figure 7.3-10c for a schematic representation of the ADS initiation logic.

Each channel contains a single input from a drywell high pressure sensor. In addition, one channel includes two differential pressure sensor inputs monitoring reactor vessel low water level (Trip Level 3 and Trip Level 1). The second low water level trip (Trip Level 3) provides confirmation of a reactor vessel low water level condition. The other channel, in addition to drywell high pressure, includes a single reactor vessel low water level (Trip Level 1) input.

To assure that adequate makeup water is available after the vessel has been depressurized each logic channel includes a pump discharge presure permissive signal indicating LPCI or LPCS system available for vessel water makeup. Any one of the three LPCI pumps or the LPCS pump is sufficient to permit automatic depressurization.

After receipt of the initiation signals and after a delay provided by timers, each of the two solenoid pilot air valves are energized. This allows pneumatic pressure from the accumulator to act on the air cylinder operator. Each ADS trip system timer can be reset manually to delay system initiation. If reactor vessel water level is restored by HPCS prior to the end of the time delay, ADS initiation will be prevented.

The ADS trip system A actuates the "A" solenoid pilot valve on each ADS relief valve. Similarly, the ADS trip system B actuates the "B" solenoid pilot valve on each ADS relief valve. Actuation of either solenoid pilot valve causes the ADS valve to open to provide depressurization.

Once initiated the ADS logic seals-in and can be reset by the control room operator only when either drywell pressure or vessel water level return to normal.

Two control switches (one for each trip system solenoid) are located in the main control room for each safety-relief valve associated with the ADS. Each switch controls one of the two solenoid pilot valves.

7.3.1.1.1.3 Low Pressure Core Spray (LPCS) - Instrumentation and Controls

a. LPCS Function

The purpose of the LPCS is to provide low pressure reactor vessel core spray following a loss-of-coolant accident when the vessel has been depressurized and vessel water level has not been restored by the HPCS. The LPCS is functionally diverse to the LPCI mode of the residual heat removal system. See 6.3.2.2.3.

b. LPCS Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 7.3-11 (LPCS P&ID). LPCS component control logic is shown in Figure 7.3-12 (LPCS FCD). Instrument specifications are listed in Tables 7.3-5 and 7.3-6. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 7.3-11 (LPCS P&ID) and Figure 7.3-12 (LPCS FCD).

The LPCS is initiated automatically by either reactor vessel low water level and/or drywell high pressure. The system is designed to operate automatically for at least 10 minutes without any actions required by the control room operator. Once initiated the LPCS logic seals-in and can be reset by the control room operator only when the initial conditions return to normal. Refer to Figure 7.3-12 for a schematic representation of the LPCS system initiation logic.

Reactor vessel water level (Trip Level 1) is monitored by two redundant differential pressure switches. To provide diversity drywell pressure is monitored by two redundant pressure switches. The vessel level switch contacts and the drywell pressure switch contacts are connected in a one-out-of-two twice logic arrangement so that no single instrument failure can prevent initiation of LPCS.

The LPCS components respond to an automatic initiation signal simultaneously (or sequentially as noted) as follows:

1. The Division 1 Diesel Generator is signalled to start,

2. The normally closed test return line to the suppression pool valve MO F012 is signalled closed,

3. When power (offsite or onsite) is available at the LPCS pump motor bus the LPCS pump is signalled to start,

b. SSW System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 9.1-4 (SSW Flow Diag.). SSW component control logic is shown in Figure 7.3-17 (SSW Control Logic Diag.). Instrument specifications are listed in Tables 7.3-21 and 7.3-22. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 9.1-4 (SSW Flow Diag.) and Figure 7.3-17 (SSW Control Logic Diag.).

The SSW system is automatically initiated as follows:

1. The Division 1 SSW pump P-1A is started automatically if water level in the spray pond is sufficient and when either the RHR A pump, the LPCS pump, or the Division 1 diesel generator is started,

2. The Division 2 SSW pump P-1B is started automatically if water level in the spray pond is sufficient and when either the RHR B pump, RHR C pump, the Division 2 diesel generator, or the RCIC pump is started,

3. The HPCS service water pump C002 is automatically started when the HPCS pump is started.

Once the service water pumps are started the following occurs:

1. The RHR heat exchanger service water discharge valves MO F068A,B are signalled open,

2. After the SSW pumps discharge pressure exceeds a minimum value the pump discharge valves V-2A,2B and V-29 are signalled to open,

The SSW pumps are automatically tripped if spray pond water level becomes too low.

7.3.1.1.7 Main Control Room and Critical Switchgear Rooms HVAC System - Instrumentation and Controls

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-19 (HVAC Flow Diag.). Component control logic is shown in Figure 7.3-14 (HVAC Control Logic Diag.). Instrument specifications are listed in Tables 7.3-19 and 7.3-20. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-19

(HVAC Flow Diag.) and Figure 7.3-14 (HVAC Control Logic Diagram).

For a complete description of the Main Control Room and Critical Switchgear Rooms HVAC Instrumentation and Controls refer to 9.4.1.

7.3.1.1.8 Containment Atmosphere Control (CAC) System - Instrumentation and Controls

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-17 (CAC Flow Diag.). CAC component control logic is shown in Figure 7.3-21 (CAC Control Logic Diag.). Instrument specifications are listed in Tables 7.3-17 and 7.3-18. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-17 (CAC Flow Diag.) and Figure 7.3-21 (CAC Control Logic Diag.).

For a complete description of the CAC System Instrumentation and Controls refer to 6.2.5.

7.3.1.1.9 Standby Gas Treatment System (SGTS) - Instrumentation and Controls

For a complete description of the SGTS Instrumentation and Controls refer to 6.5.1.

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-16 (SGTS Flow Diag.). SGTS component control logic is shown in Figure 7.3-19 (SGTS Control Logic Diag.). Instrument specifications are listed in Tables 7.3-15 and 7.3-16. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-16 (SGTS Flow Diag.) and Figure 7.3-19 (SGTS Control Logic Diag.).

7.3.1.1.10 Reactor Building Ventilation and Pressure Control System - Instrumentation and Controls

a. System Function

The reactor building ventilation and pressure control system automatically maintains subatmospheric pressure of 1/4" water gage in the reactor building atmosphere. See 9.4.2.

b. System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-18 (HVAC React. Bldg. Flow Diag.). System component control logic is shown in Figure 7.3-19 (SGTS Control

Logic Diag.). Instrument specifications are listed in Tables 7.3-15 and 7.3-16. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-18 (HVAC React. Bldg. Flow Diag.) and Figure 7.3-19 (SGTS Control Logic Diag.).

The differential pressure is monitored by eight redundant differential pressure transmitters, four in Division 1 and four in Division 2, which measure the differential pressure from the exterior of four sides of the reactor building to the fuel pool area. The signal indicating the least differential pressure from the four differential pressure transmitters in one division is selected and is used to control the position of a damper of that division in the normal reactor building exhaust fan units or upon the initiation of the standby gas treatment system by containment isolation signals high drywell pressure, low reactor water level, or reactor building exhaust high radiation, the reactor building ventilation and pressure control system then controls reactor building pressure by controlling the standby gas treatment system fan units. (See 6.5.1).

7.3.1.1.11 Containment Instrument Air (CIA) System

a. CIA System Function

The purpose of the containment instrument air system is to provide uninterruptable divisional instrument air to essential ADS valve accumulators inside primary containment and non-safety-related instrument air supplies to other valves inside containment as shown in Figure 3.2-21. The system consists of a safety-related portion, which is comprised of two divisions, and a non-safety-related portion. During normal operation, the non-safety-related portion of the system provides control air to ADS accumulators and other valves.

In the event of failure of the non-safety-related portions of the system, which is indicated by low header pressure and detected by two redundant pressure switches, the safety-related portion automatically maintains header pressure from two nitrogen bottle sources. The non-safety-related portion of the system is isolated from the safety-related portion upon detection of failure of the non-safety-related portion. See 9.3.1.2.2.

b. CIA System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-21 (CIA Flow Diag.). CIA component control logic is shown in Figure 7.3-20 (CIA Control Logic Diag.). Instrument specifications are listed in Tables 7.3-13 and

7.3-14. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-21 (CIA Flow Diag.) and Figure 7.3-20 (CIA Control Logic Diag.).

The containment instrument air system is always in operation. The instrumentation and controls of the system perform the following functions:

1. Monitor CIA system header pressure

2. Monitor CIA system compressor operation

3. Isolate the non-safety-related portion of system in the event of failure in this portion

4. Maintain CIA system header pressure in the event of item 3 by sequentially opening nitrogen bottles.

7.3.1.2 Design Basis

The ESF systems are designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are presented in that chapter.

a. Variables Monitored to Provide Protective Action

The following variables are monitored in order to provide protective actions to the ESF systems:

1. HPCS

   a) Reactor Vessel Low Water Level (Trip Level 2)

   b) Drywell High Pressure

2. ADS

   a) Reactor Vessel Low Water Level (Trip Level 3)

TABLE 7.3-28

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR MAIN STEAM LINE LEAKAGE CONTROL SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Pressure Low (MSLC-PS 20) (MSLC-PS 8A-D)	5	1
MSLC Header Press. Low (MSLC-PS 25)	1	0
MSLC Header Press. Low (MSLC-PS 70A-D)	4	0
MSLC High Flow	4	0

BLANK

Accident Conditions

Information readouts are designed to accommodate all credible accidents for operator actions, information, and event tracking requirements, and cover all other design basis events or incident requirements.

Regulatory Guide 1.97, Rev. 2 is scheduled to be issued in August 1980. The Guide will provide direction to near-term OL plants (WNP-2) for implementation of specific requirements (quality, single failure, power source, etc.) as they apply to safety-related display instrumentation.

WNP-2 has responded to Regulatory Guide 1.97, draft 2 in letter G02-80-29, D. L. Renberger (WPPSS) to H. Denton (NRC) dated Feb. 1, 1980. The response included comments on the guide as well as a detailed description of the WNP-2 design.

A complete analysis will be provided for safety-related display instrumentation following issuance of Regulatory Guide 1.97, Rev. 2.

TABLE 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

Design Criteria	Type Readout	Number of Channels	Range	Accuracy	Location
Reactor Vessel Pressure	Recorder	2	0-1,500 psig	± 2% FS	CR
Reactor Vessel Water Level	Recorder	2	-150"/0/+60"	± 2% FS	CR
RCIC Flow	Meter	1	0-700 GPM	± 1% FS	CR
RCIC Discharge Pressure	Meter	1	0-1500 psig	± 2% FS	CR
HPCS Flow	Meter	1	0-10,000 GPM	± 2% FS	CR
HPCS Discharge Pressure	Meter	1	0-1,500 psig	± 2% FS	CR
LPCS Flow	Meter	1	0-10,000 GPM	± 2% FS	CR
RHR Flow (LPCI and Shut-down Cooling)	Meter	3	0-10,000 GPM	± 2% FS	CR
RHR Service Water Flow	Meter	2	0-10,000 GPM	± 2% FS	CR
Drywell Pressure	Recorder	2 2	0-2 psig LOW 2 0-100 psig HIGH	± 2% FS	CR
MSIV-LCS Outboard Steam Line Header Pressure	Meter	1	0-1200 psig	± 2% FS	CR
MSIV-LCS Steam Line Pressure Between MSIV's	Meter	4	0-1200 psig	± 2% FS	CR
MSIV/LCS Leakage Flow	Meter	4	0-0.5 CFM	+ 2% FS	CR

TABLE 7.5-1 (Continued)

Design Criteria	Type Readout	Number of Channels	Range	Accuracy	Location
SSW System Pump Discharge Line Pressure	Meter	2 1	0-300 psig 0-100 psig	± 1% FS ± 1% FS	CR CR
Suppression Pool Water Level	Recorder	2	-25"-0-+25"	± 1%	CR
Main Control Room Temperature	Meter	2	50-100°F	± 2%	CR
SGTS Flow Rate	Meter	4	0-6000 CFM	± 3%	CR
CAC System Flow Rate	Meter	4	0-300 CFM	± 3%	CR
Primary Containment Hydrogen	Recorder	2	0-10% H_2	± 1%	CR
Suppression Pool Water Temperature	Recorder	2	50-400°F	± 1%	CR

TABLE 7.5-2

CONTAINMENT HYDROGEN AND OXYGEN MONITORING SYSTEM SAMPLE POINT LOCATIONS

SP	PENETRATION #	SAMPLE POINT AZIMUTH	SAMPLE POINT ELEVATION
74	72c	188°-24'	560'-0"
75	72d	191°-36'	560'-0"
76	72e	193°	531'-0"
77	82c	230°	479'-4"
78	85d	18°-12'	545'-2-1/4"
79	85e	13°-44'	545'-1-1/2"
80	73d	45°	531'-0"
81	84b	40°	479°-4"

7.6 ALL OTHER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

7.6.1 DESCRIPTION

Section 7.6 describes the instrumentation and control systems required for safety not discussed in other sections. The systems include:

1. Process Radiation Monitoring System
2. High Pressure/Low Pressure Systems Interlocks
3. Leak Detection System (LDS)
4. Neutron Monitoring System (NMS)
5. Recirculation Pump Trip System (RPT)
6. Spent Fuel Pool Cooling and Cleanup System
7. Suppression Pool Temperature Monitoring System

The sources which supply power to the safety-related systems described in this section originate from on-site ac and/or dc safety-related busses or as in the case of the fail-safe logic NMS and portions of the LDS, from the non-safety-related RPS MG-sets. Refer to Chapter 8 for a complete description of the safety-related systems power sources.

7.6.1.1 Process Radiation Monitoring System - Instrumentation and Controls

The safety-related portions of the process radiation monitoring system are described in 7.2.1.1.b and 7.3.1.1.2.b.

7.6.1.2 High Pressure/Low Pressure Systems Interlocks

a. System Function

Instrumentation and controls are provided to prevent overpressurization of low pressure systems which interface with the reactor coolant pressure boundary.

b. System Operation

Schematic arrangement of mechanical equipment for the systems involved is shown in Figures 7.3-13 (RHR P&ID) and 7.3-11 (LPCS P&ID). Component control logic for the systems involved is shown in Figures 7.3-14 (RHR FCD) and 7.3-12 (LPCS FCD). Electrical schematics are identified in 1.7. Instrument specifications are listed in Tables 7.6-5 and 7.6-6.

The following high pressure/low pressure interlock equipment is provided:

Interlocked Process Line	Type	Valve	Parameter Sensed	Purpose
RHRS Shutdown Supply	MO MO	F009 F008	Reactor pressure	Prevents valve opening until reactor pressure is below system design pressure
RHRS Shutdown Return	MO	F053	Reactor Pressure	Prevents valve opening until reactor pressure is below system design pressure
RHRS Head Spray	MO	FO23	Reactor Pressure	Prevents valve opening until reactor pressure is below system design pressure
LPCI	MO	FO42	Differential pressure across valve	Does not allow valve to open until differential pressure is low
LPCS	MO	F005	Differential pressure across valve	Does not allow valve to open until differential pressure is low

The shutdown cooling suction valves, head spray valve, and discharge valve have redundant and diverse interlocks to prevent the valves from being opened when the primary system pressure is above the subsystem design pressure. These valves also receive a signal to close when reactor pressure is above system pressure.

The LPCI and LPCS discharge valves MO F042 and MO F005 are prevented from opening until differential pressure across the valves is low enough to prevent system overpressurization.

7.6.1.3 Leak Detection System - Instrumentation and Controls

The safety-related portions of the Leak Detection System are as follows:

1. Main Steam Line Leak Detection
2. RCIC System Leak Detection

In addition, wall-mounted level switches are provided in each sump room to detect ECCS passive failures and provide annunciation in the main control room.

7.6.1.4 Neutron Monitoring System (NMS) - Instrumentation and Controls

The safety-related portions of the neutron monitoring system are as follows:

1. Intermediate Range Monitor (IRM)

2. Local Power Range Monitor (LPRM)

3. Average Power Range Monitor (APRM)

a. Neutron Monitoring System Function

The neutron monitoring system instrumentation and controls are designed to monitor reactor power (neutron flux) from startup through full power operation.

b. Neutron Monitoring System Operation

The neutron monitoring system uses incore detectors, either fixed (LPRM) or removable (IRM), to determine neutron flux levels.

NMS will initiate a scram when predetermined limits are exceeded and provide operator information during and after accident conditions.

The NMS component control logic is shown in Figure 7.6-6.

7.6.1.4.1 Intermediate Range Monitor (IRM)

a. IRM Function

The IRM monitors neutron flux from the upper portion of the SRM range to the lower portion of the power range (APRM) as shown in Figure 7.6-22.

b. IRM Operation

The IRM has eight channels, each of which includes one detector that can be positioned in the core by remote control. Refer to Figures 7.6-5 and 7.6-2. The detectors are inserted into the core for a reactor startup and are withdrawn after the reactor mode selector switch is placed in the RUN position.

Each detector assembly consists of a fission chamber attached to a low-loss, quartz-fiber-insulated transmission cable. When coupled to the signal conditioning equipment, the detector produces a reading of full scale on the most sensitive range. The detector cable is connected underneath the reactor vessel to a triple-shielded cable that is connected to the preamplifier.

The preamplifier converts current pulses to voltage pulses, modifies the voltage signal, and provides impedance matching. The preamplifier output signal is then sent to the IRM signal conditioning electronics (see Figure 7.6-8).

Each IRM channel input signal from the preamplifier can be amplified and attenuated. IRM preamplification is selected by a remote range switch that provides 10 ranges of increasing attenuation (The first 6 called low range and the last 4 called high range). As the neutron flux of the reactor core increases the signal from the fission chamber is attenuated to keep the input signal to the inverter in the same range. The output signal, which is proportional to neutron flux at the detector, is amplified and supplied to a locally mounted meter, a remote meter and recorder.

The IRM Scram Trip Functions are discussed in 7.2.1.1.B. The IRM trips are shown in Table 7.6-1.

The IRM range switches must be up-ranged or down-ranged to follow increases and decreases in power within the range of the IRM to prevent either a scram or a rod block. The IRM detectors must be inserted into the core whenever these channels are needed, and withdrawn from the core, when permitted, to prevent unnecessary burnup.

7.6.1.4.2 Local Power Range Monitor (LPRM)

a. LPRM Function

The LPRM's provide localized neutron flux detection over the full power range for input to the APRM.

b. LPRM Operation

The LPRM includes 43 detector strings having detectors located at different axial heights in the core; each detector string contains four fission chambers. Figure 7.6-3 shows the LPRM detector radial layout scheme.

b. APRM Operation

The APRM has six redundant channels. Each channel uses input signals from a number of LPRM channels. Three APRM channels are associated with each trip system of the RPS.

The APRM channel uses electronic equipment that averages the output signals from a selected set of LPRMs, trip units that actuate automatic devices, and signal readout equipment. Each APRM channel can average the output signals from as many as 24 LPRMs. Assignment of LPRMs to an APRM follows the pattern shown in Figure 7.6-4. Position A is the bottom position, Positions B and C are above Position A, and Position D is the topmost LPRM detector position. The pattern provides LPRM signals from all four core axial LPRM detector positions.

The APRM amplifier gain can be adjusted by combining fixed resistors and potentiometers to allow calibration. The averaging circuit automatically corrects for the number of unbypassed LPRM amplifiers providing inputs to the APRM.

Refer to 7.2.1.1.b for a description of the APRM inputs to the RPS.

APRM system trips are summarized in Table 7.6-3. The APRM circuit arrangment for RPS trip input is shown in Figure 7.6-10.

One of the two recirculation flow signals in each trip system may be bypassed at any time. One of the three APRMs in each trip system may be bypassed at any time. An interlock circuit provides an inoperative trip output from an APRM whenever the minimum number of LPRM inputs to it is not met.

The APRM channels receive power from the 120 Vac RPS MG sets. Power for each APRM trip unit is supplied from the same power supply as the APRM it services. APRM Channels A, C, and E are powered from the bus used for Trip System A of the RPS; APRM Channels B, D, and F are powered from the bus used for RPS Trip System B. The ac bus used for a given APRM channel also supplies power to its associated LPRMs.

7.6.1.5 Recirculation Pump Trip (RPT) System - Instrumentation and Controls

See Appendix H and 5.4 for a description of the RPT system.

7.6.1.6 Spent Fuel Pool Cooling and Cleanup System (FPC) - Instrumentation and Control

a. FPC System Function

The function of the FPC system is to remove decay heat from the spent fuel storage pool to insure adequate cooling of irradiated stored fuel assemblies. The FPC system also purifies the storage pool water, maintains water clairty for fuel handling operations, and fills and drains the fuel transfer canal. Refer to 9.1.3.

b. FPC System Operation

Schematic arrangement of the FPC system mechanical equipment is shown in Figure 3.2-12 (FPC P&ID). FPC system component control logic is shown in Figure 7.6-11 (FPC Logic Diag.). Instrument Specifications are listed in Tables 7.6-11 and 7.6-12. Plant layout drawings and electrical schematics are shown in 1.7. Operator information displays are shown in Figure 3.2-12 (FPC P&ID) and Figure 7.6-11 (FPC Logic Diag.)

The FPC System consists of two redundant cooling loops. The system is manually initiated and one loop runs continuously when the pool contains spent fuel.

Instrumentation is provided to monitor the pool temperature, pump suction and discharge pressures, and water conductivity to allow the control room operator to assess system operation.

7.6.1.7 Suppression Pool Temperature Monitoring System - Instrumentation and Controls

a. System Function

The suppression pool temperature monitoring (SPTM) system is designed to monitor suppression pool water temperature and alert the plant operator to the potentially hazardous condition of elevated pool water temperature.

The instrumentation for the SPTM system is shown in Figure 3.2-8.

b. System Operation

The suppression pool temperature monitoring system consists of 24 dual element thermocouples. Sixteen thermocouples are arranged near the surface of the pool whereas the remaining 8 are located midlevel in the pool. This arrangement was chosen to track pool stratification. The sensors are separated into two redundant divisions and maintained throughout the system.

The time constant for the thermocouples is no greater than 15 seconds. The time from signal output of sensor to initiation of alarm is no greater than 0.5 seconds. The difference between measurement reading and actual temperature is within $\pm 2°F$.

Each division of the Suppression Pool Temperature Monitoring System is provided with temperature readout devices in the main control room consisting of four indicators and a four channel recorder. Each division is provided with an audible and visual annunciator in the control rool.

7.6.1.8 Design Basis

The safety-related systems described in 7.6 are designed to provide timely protective action inputs to other safety systems to protect against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, Appendix 15A, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are also presented in Chapter 15.

The station conditions which require protective actions are described in Chapter 15 and Appendix 15A.

a. Variables Monitored to Provide Protective Actions

The following variables are monitored in order to provide protective action inputs:

1. High Pressure/Low Pressure System Interlocks
   a) Reactor pressure
   b) Differential pressure across the LPCS and LPCI injection valves
2. Leak Detection System
   a) RCIC area temperatures - differential and ambient
   b) RCIC steam line flow rate
   c) RCIC turbine exhaust diaphragm pressure

d) RCIC steam line pressure

e) RHR area temperatures - differential and ambient

f) RHR shutdown cooling suction flow

g) RWCU area temperatures - differential and ambient

h) RWCU differential flow

i) Identified and Unidentified leakage from the drywell floor and equipment drain sumps.

3. Neutron Monitoring System

a) IRM neutron flux

b) APRM neutron flux

4. Spent Fuel Pool Cooling and Cleanup System

a) Fuel Pool Temperature

5. Suppression Pool and Chamber Temperature Monitoring System

a) Suppression Pool and Chamber Temperature

The plant conditions which require protective action involving the safety-related systems discussed in 7.6 are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

See Chapter 16 for the minimum number of sensors required to monitor safety-related variables. The IRM and LPRM detectors are the only sensors which have spatial dependence.

c. Prudent Operational Limits

Prudent operational limits for each safety-related variable trip setting are selected to be far enough above or below normal operating levels so that a spurious safety system initiation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or nuclear system process barrier, is kept within acceptable bounds.

TABLE 7.6-7 (Continued)

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Main Steam Line Steam High Flow	Diff. Press. Switch (E31-N008A-D thru E31-N011A-D)	-15 to 0 to 150 psid	104 psid	-	-	-
RWCU Diff. Flow High	Diff. Flow Switch (E31-N605A,B)	-	57 gpm	-	-	-
React. Bldg. Equip. Draw Sump Level High	Level Switch (EDR-LS-N014)	-	-	-	-	-
React. Bldg. Floor Drain Sumps Level High	Level Switch (FDR-LS-N006A, B) (FDR-LS-N005A, B)	-	-	-	-	-

NOTES FOR TABLE 7.6-7

(1) See Chapter 16, "Technical Specifications" for operational limits.

The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored.

(2) Trip settings shown are subject to change to comply with Chapter 16, "Technical Specifications".

The trip setpoint is located in that portion of an instrument's range which provides the required accuracy.

Initial trip setting values are established from operating experience with similar size plants, and back up with analysis as necessary.

(3) See Chapter 16, "Technical Specifications" for instrument setpoint margins.

(4) Values shown are subject to change to comply with Chapter 16, "Technical Specifications".

TABLE 7.6-8

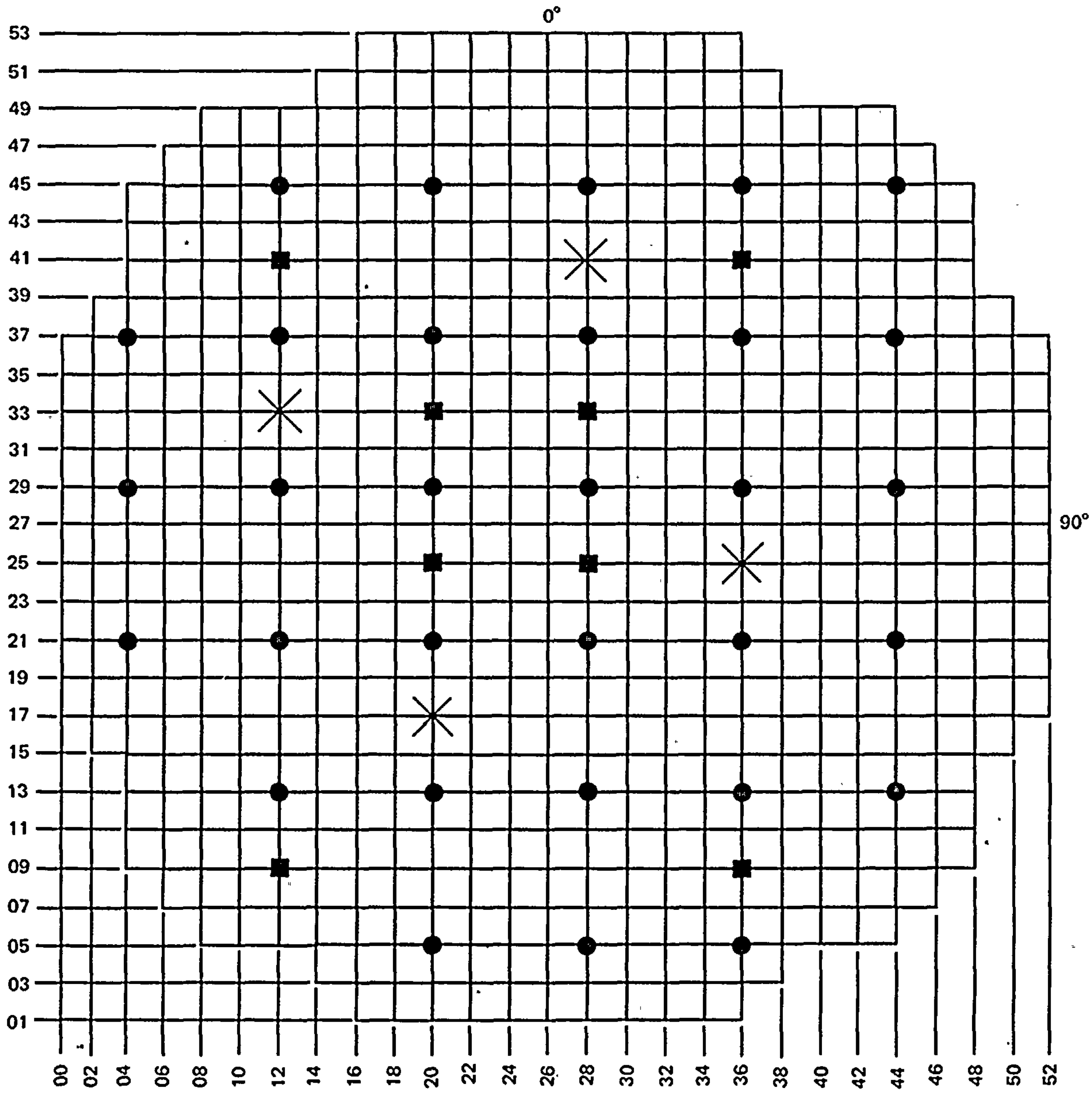
CHANNELS REQUIRED FOR PROTECTIVE ACTION COMPLETION FOR THE LEAK DETECTION SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
RCIC Steam Supply Press. Low	4	1
RCIC Steam Supply High Flow or Instrument Line Break	4	1
RCIC Turbine Exhaust Press. High	4	1
RCIC Equip. Area High Diff. Temp.	2	1
RCIC Pipe Routing Area High Diff. Temp.	2	1
RWCU Equip. Area High Diff. Temp.	6	1
RWCU Equip. Area Ambient Temp. High	6	1
RCIC Equip. Area Ambient Temp. High	2	1
RCIC Pipe Routing Area Diff. Temp.	2	1
Main Steam Line Tunnel Temp. High	4	1
Main Steam Line Tunnel Diff. Temp. High	4	1
RHR Area Ambient Temp. High	2	1
RHR Equip. Area Diff. Temp. High	2	1

TABLE 7.6-8 (continued)

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Main Steam Line Pipe Routing Area in Turbine Gen. Bldg. Ambient Temp. High	48	2
Main Steam Line Steam High Flow	16	2
RWCU Diff. Flow High	2	1
Reactor Bldg. Equip. Drain Sump Level High	1	1
Reactor Bldg. Floor Drain Sumps Level High	4	1

AMENDMENT NO. 10
July 1980



●	LOCAL POWER RANGE MONITORING SYSTEM (LRPM)	31
✳	SOURCE RANGE MONITORING SYSTEM (SRM)	4
■	INTERMEDIATE RANGE MONITORING SYSTEM (IRM)	8
	TOTAL PENETRATIONS FOR NUCLEAR INSTRUMENT	43

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	VESSEL PENETRATIONS FOR NUCLEAR INSTRUMENTATION.	FIGURE 7.6-3

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 DESCRIPTION

Section 7.7 describes instrumentation and controls of major plant control systems whose functions are not essential for the safety of the plant. This section also describes instrumentation and controls, not essential for the safety of the plant, which are not discussed in any other FSAR section. The systems include:

1. Reactor Vessel Instrumentation
2. Reactor Manual Control System (RMCS)
3. Recirculation Flow Control System
4. Feedwater Control System
5. Pressure Regulator and Turbine - Generator System
6. Neutron Monitoring System (TIP, SRM, RBM)
7. Process Computer System and Rod Worth Minimizer Function (RWM)
8. Rod Sequence Control System (RSCS)
9. Loose Parts Detection System (LPDS)
10. Refueling Interlocks

Refer to Tables 7.7-1 and 7.7-2 for system design and supply responsibility and similarity to licensed reactors, respectively.

7.7.1.1 Reactor Vessel - Instrumentation

Figure 7.3-9 (Nuclear Boiler System P&ID) shows the arrangements of the sensors, and sensing equipment used to monitor reactor vessel conditions.

a. System Function

The purpose of the reactor vessel instrumentation is to monitor key reactor vessel variables to provide the operator with information during normal plant operation, startup and shutdown.

b. System Operation

The following is a discussion of each reactor vessel variable monitored:

1. Reactor Vessel Temperature

The reactor vessel temperature is determined on the basis of reactor coolant temperature. Temperatures needed for operation and for compliance with the Technical Specification operating limits are obtained from one of several sources, depending on the operating condition. During normal operation, either reactor pressure and/or the inlet temperature of the coolant in the recirculation loops can be used to determine the vessel temperature. Below the operating span of the temperature sensors (RTD's) in the recirculation loop and above 212°F, the vessel pressure is used for determining the temperature. Below 212°F, the vessel coolant, and thus the vessel temperature, is determined by the reactor water cleanup system inlet temperature. During normal operation, vessel thermal transients are limited via operational constraints on parameters other than temperature.

2. Reactor Vessel Water Level

Figure 7.7-1 shows the water level range and the reactor vessel tap location for each water level range. The instruments that sense the water level are differential pressure devices with a condensate reference leg, calibrated to be accurate at a specific vessel pressure and liquid temperature condition. The following is a description of each water level range shown on Figure 7.7-1:

a) Shutdown Water Level Range: This range is used to monitor the reactor water level during reactor shutdown conditions when the reactor system is flooded for maintenance and head removal. The vessel temperature and pressure condition that is used for the calibration is 0 psig and 120°F water in the vessel. The two vessel instrument tap elevations used for this water level measurement are located at the top of the reactor vessel head and the instrument tap just below the bottom of the dryer skirt.

b) Upset Water Level Range: This range is used to monitor the reactor water level above the narrow range scale (see c.

This system is designed to meet the environmental conditions described in Table 3.11-4 for the control room.

The RSCS is designed primarily to mitigate the consequences of the postulated rod drop accident, which analysis shows to be of no concern at power levels in excess of 20% rated thermal power. Mitigation is achieved by constraining control rod movements by the operator to pre-determined patterns and sequences which ensure that control rods of high worth are not obtained below the 20% power level. The design criterion is that any potential rod drop accident should not result in fuel rod enthalpies in excess of 280 cal/g. Over the operating ranges of power level and fuel exposure, the resultant fuel rod enthalpy is a function of several parameters, of which control rod worth is the most significant and controllable.

To meet the 20% rated power level requirement, the RSCS is required to be in operation during reactor startup and shut-down between 0 and a nominal 30% rated thermal power. The RSCS function is supported by the redundant action of the Rod Worth Minimizer (RWM), which is programmed to permit only the same rod patterns and sequences as prescribed for the RSCS. While the startup or shutdown of the reactor may continue without the RWM when a second licensed operator is present to check rod movements, the RSCS must be operable at all times over the prescribed power range (0-30%). For example, the reactor may not be started up (rods pulled) if the RSCS is inoperable, and if the RSCS fails during startup or shutdown (0-30%) power, rods may be moved only by scramming the reactor.

Control rod drives may have to be declared inoperable for various reasons as defined in the technical specifications, and this in turn leads to restrictions on the position and use of such rods. The RSCS is designed to be compatible with these restrictions and has facilities for the bypassing of inoperable control rods.

7.7.1.12 Loose Parts Detection System (LPDS) - Instrumentation and Controls

a. LPDS Function

The loose parts detection system monitors the reactor vessel for the presence of internal loose parts. Internal movement of components or core vibration of internals is not required to be monitored by this equipment.

b. LPDS Operation

The loose parts detection sensors are mounted on the exterior of the primary coolant system and located at natural collection points where loose parts will most likely impact. During startup and normal plant operation the LPDS is on line to provide visual and audio information to the operator. Also, this data is recorded on magnetic tape for detailed analysis at a later date and to provide a startup base line signature of various pumps and valves to be used for comparative analysis. LPDS schematic arrangements are shown in Figure 7.7-15 (LPDS P&ID).

7.7.1.13 Refueling Interlocks - Instrumentation and Controls

a. Refueling Interlocks Function

The purpose of the refueling interlocks is to restrict the movement of control rods and the operation of refueling equipment. This rein-forces operational procedures that prevent the reactor from becoming critical during refueling operations.

b. Refueling Interlocks Operation

The refueling interlocks circuitry senses the condition of the refueling equipment and the control rods to prevent the movement of the refueling equipment or withdrawal of control rods (rod block). Redundant circuitry is provided to sense the following conditions:

1. All rods inserted

2. Refueling platform positioned near or over the core

3. Refueling platform hoists fuel-loaded (fuel grapple, frame-mounted hoist, trolley-mounted hoist)

4. Service platform hoist fuel-loaded, and

5. Reactor Mode Switch in "Refuel" position.

The indicated conditions are combined in logic circuits to satisfy all restrictions on refueling equipment operations and control rod movement (Figure 7.7-3). A two-channel circuit indicates that all rods are in. The rod-in condition for each rod is established by the closure of a magnetically operated reed switch in the rod position indicator probe. The rod-in

switch must be closed for each rod before the all-rods-in signal is generated. Both channels must indicate "all-rods-in" to allow refueling equipment to be used.

During refueling operations, no more than one control rod is permitted to be withdrawn; this is enforced by a redundant logic circuit that uses the all-rods-in signal and a rod selection signal from the reactor manual control system to prevent the selection of a second rod for movement with any other rod not fully inserted. Control rod withdrawal is prevented by comparison between the A and B portions of the RMCS for rod position with a subsequent rod withdrawal block if necessary. The simultaneous selection of two control rods is prevented by the interconnection arrangement of the select pushbuttons. With the mode switch in the REFUEL position, the circuitry prevents the withdrawal of more than one control rod and the movement of the loaded refueling platform over the core with any control rod withdrawn.

Operation of refueling equipment is prevented by interrupting the power supply to the equipment. The refueling platform is provided with two mechanical switches attached to the platform, which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. The switches open before the platform or any of its hoists are physically located over the reactor vessel to indicate the approach of the platform toward its position over the core.

Load cell readout is provided for all hoists. Indicators display given hoist loads directly to the operator. Load sensing is by hydraulic load cells that use demineralized water as the operating fluid. Associated interlock and load functions are performed by pressure switches that sense the pressure generated by the hydraulic load cells.

The three hoists on the refueling platform and the hoist on the service platform are provided with switches that open when the hoists are fuel loaded. The switches open at a load weight that is lighter than that of a single fuel assembly. This indicates when fuel is loaded on any hoist.

A bypass for the service platform hoist load interlock is provided. When the service platform is no longer needed, its power plug is removed. This de-energizes the power supply to the hoist. The platform can then be moved away from the core.

De-energizing the hoist power supply opens the hoist load switches and gives a false indication that the hoist is loaded. This indication prevents control rod withdrawal with the mode switch in the STARTUP or REFUEL positions. A bypass

plug allows control rod movement in this situation. The bypass plug is physically arranged to prevent the connection of the service platform power plug unless the bypass plug is removed.

The rod block interlocks and refueling platform interlocks provide two independent levels of interlock action. The interlocks which restrict operation of the platform hoist and grapple provide a third level of interlock action since they would be required only after a failure of a rod block and refueling platform interlock.

In the refueling mode, the control room operator has an indicator light for "Refueling Mode Select Permissive" whenever all control rods are fully inserted. He can compare this indication with control rod position data from the computer as well as control rod in-out status on the full core status display. Whenever a control rod withdrawal block situtation occurs, the operator receives annunciation and computer logs of the rod block. The operator can compare these outputs with the status of the variable providing the rod block condition. Both channels of the control rod withdrawal interlocks must agree that permissive conditions exist in order to move control rods; otherwise, a control rod withdrawal block occurs. Failure of one channel may initiate a rod withdrawal block, and will not prevent application of a valid control rod withdrawal block from the remaining operable channel (see Table 7.7-3).

In terms of refueling platform interlocks, the platform operator has analog type readout indicators for the platform x-y position relative to the reactor core.

The position of the grapple is shown in a digital indicator immediately below the platform position indicators. Analog load cell indications of hoist loads are given for each hoist by locally mounted indicators. Individual pushbutton and rotary control switches are provided for local control of the platform and its hoists. The platform operator can immediately determine whether the platform and hoists are responding to his local instructions, and can, in conjunction with the control room operator, verify proper operation of each of the three categories of interlocks listed previously.

7.7.1.14 Design Differences

Refer to Table 7.7-2 for a list of Instrumentation and Control system designs and their similarity to designs of other nuclear power plants.

7.7.2 ANALYSIS

Refer to the safety evaluations in Chapter 15 and Appendix A of Chapter 15. Chapter 15 shows that the systems described in 7.7 are not utilized to provide any design basis accident safety function. Safety functions are provided by other systems.

Chapter 15 also evaluates all credible control system failure modes, the effects of those failures on plant functions, and the response of various safety-related systems to those failures.

The major plant control systems described above have no direct interface with any safety-related systems and, thus, control system failures, other than those described in Chapter 15, have no effect on the safety-related systems.

TABLE 7.7-1

DESIGN AND SUPPLY RESPONSIBILITY OF PLANT CONTROL SYSTEMS

	GE DESIGN	GE SUPPLY	B&R DESIGN	OTHERS SUPPLY
Reactor Vessel Instrumentation	X	X		
Reactor Manual Control System	X	X		
Recirculation Flow Control System	X	X		
Feedwater Control System	X	X	X	X
Press. Regulator & Turbine Generator System			X	X
Neutron Monitoring System				
SRM	X	X		
RBM	X	X		
TIP	X	X		
Process Comp. & RWM	X	X		
Rod Sequence Control System	X	X		
Loose Parts Detection System			X	X
Refueling Interlocks	X	X		

TABLE 7.7-2

SIMILARITY TO LICENSED REACTORS

Instrumentation and Controls (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(1) Neutron Monitoring System (TIP, SRM, RBM)	LaSalle	Identical
(2) Refueling Interlocks	LaSalle	Identical
(3) Reactor Manual Control System	Zimmer-1	Identical
(4) Reactor Vessel - Instrumentation	Zimmer-1	Identical
(5) Recirculation Flow-Control System	Zimmer-1	Identical
(6) Feedwater Control System	Zimmer-1	Identical
(7) Pressure Regulator and Turbine-Generator System	Zimmer-1	Identical
(8) Rod Sequence Control System	Zimmer-1	Identical
(9) Refueling Interlocks	Zimmer-1	Identical
(10) Process Computer (RWM)	Vermont Yankee & subsequent plants	See Note 1
(11) Loose Parts Detection System	None	--

Note 1:

A General Electric Model 4010 process computer is used on this plant instead of a model 4020, as used on Vermont Yankee. This difference in computer equipment is insignificant.

TABLE 7.7-3

REFUELING INTERLOCK EFFECTIVENESS

SITUATION	REFUELING PLATFORM POSITION	REFUELING TMH*	PLATFORM FMH*	HOISTS FG*	SERVICE PLATFORM HOIST	CONTROL RODS	MODE SWITCH	ATTEMPTS	RESULT
1.	Not near core	UL*	UL*	UL*	UL*	All rods in	Refuel	Move refueling platform over core	No restrictions
2.	Not near core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod
3.	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	No restrictions
4.	Not near core	Any hoist loaded or FG not fully up			UL	One rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core
5.	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core
6.	Over core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod
7.	Over core	Any hoist loaded or FG not fully up				All rods in	Refuel	Withdraw rods	Rod block
8.	Not near core	UL	UL	UL	L*	All rods in	Refuel	Withdraw rods	Rod block
9.	Not near core	UL	UL	UL	L	All rods in	Refuel	Operate serviced platform hoist	No restrictions
10.	Not near core	UL	UL	UL	L	One rod withdrawn	Refuel	Operate service platform hoist	Hoist operation prevented
11.	Not near core	UL	UL	UL	UL	All rods in	Startup	Move refueling platform over core	Platform stopped before over core
12.	Not near core	UL	UL	UL	L	All rods in	Startup	Operate service platform hoist	No restictions
13.	Not near core	UL	UL	UL	L	One rod withdrawn	Startup	Operate serviced platform hoist	Hoist operation prevented
14.	Not near core	UL	UL	UL	L	All rods in	Startup	Withdraw rods	Rod block
15.	Not near core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	No restrictions
16.	Over core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	Rod block
17.	Any		Any condition		Any condition	Any condition, reactor not at power	Startup	Turn mode switch to RUN	Scram

* TMH - Trolley Mounted Hoist   FMH - Frame Mounted Hoist   FG - Fuel Grapple   UL - Unloaded   L - Fuel Loaded

TABLE 8.3-3

DIVISION 3 (HPCS) DIESEL-GENERATOR LOADING SEQUENCE
AUTOMATIC AND MANUAL LOADING OF ENGINEERED SAFETY SYSTEMS BUS

	Item Description	No. On Bus	Total HP/KW Connected To Bus	SHUTDOWN WITH LOSS OF OFFSITE POWER				LOCA WITH LOSS OF OFFSITE POWER			
				No. Req'd Part Of Set	Time to Start (1)	Time to Stop	KW	No. Req'd Part Of Set	Time to Start (1)	Time to Stop	KW
1)	HPCS Pump	1	3000/2380kw	-	-	-	-	1	0 Sec	(4)	2380
2)	Motor Operated Valves (5)	Set	77/62kw	-	-	-	-	Set	0 Sec	(4)	-
3)	Standby Water Leg Pump	1	15/12kw	-	-	-	-	1	(3)	(3)	12
4)	HPCS Service Water Pump	1	60/50kw	-	-	-	-	1	45 Sec	(4)	50
5)	Auxiliary Panel	1	24kw	1	0 Sec	(3)	24	1	0 Sec	(3)	24
6)	D-G Room Exhaust Fan	1	50/40kw	-	-	-	-	1	(3)	(3)	40
7)	Supply Fans	2	45/36kw	-	-	-	-	2	(3)	(3)	36
8)	Fan Coil Unit	1	10/8kw	1	0 Sec	(3)	5	1	0 Sec	(3)	5
9)	Instrument & Control Power	Pnl	6kw	Pnl	0 Sec	(4)	2	Pnl	0 Sec	(4)	5
10)	Lighting	Pnl	12kw	Pnl	0 Sec	(4)	2	Pnl	0 Sec	(4)	3
11)	Miscellaneous Auxiliary	Set	3kw	Set	0 Sec	(4)	2	Set	0 Sec	(4)	3
12)	HPCS Batt. Chgr.	1	5kw	1	0 Sec	-	5	1	0 Sec	-	5
						Total	40kw			Total	2563kw

For Notes see bottom of Table 8.3-2

TABLE 8.3-4a

DIVISION 1 125V DC BATTERY/SYSTEM LOADS

EQUIPMENT	L1 NORMAL	L1 INRUSH	L2 0-3 SEC	L2 3-13 SEC	L2 13-30 SEC	L2 30-60 SEC	L2 60 SEC-2HR.
MC-S1-1D	69	423	423	69	47	47	0
4.16 kV & 6.9 kV Ckt. Bkrs.	4.15 CL 4.15 OP	75	75	75	75	1	1
480V Ckt. Bkrs.	4.0 CL 4.0 OP	17	17	17	17	1	1
PP-7A-A (Inverter IN-3)	81	-	54	54	54	54	54
DP-S1-1A	36	-	17	17	17	17	17
DP-S1-1B	35	-	22	22	22	22	22
DP-S1-1C[1]	28	-	20	20	20	20	20
DP-S1-1D[1]	15	-	8	8	8	8	8
DP-S1-1E[1]	64	-	64	64	64	5	5
TOTAL	-	-	700	346	324	175	128

NOTES:

1. Ckt. Bkr. close (CL) and trip (OP) loads supplied from these panels are accounted for in Ckt. Bkr. items above.
2. L1 = Maximum connected load (LOCA with loss of offsite power).
   L2 = System accident load (LOCA with loss of offsite power).
3. Normal system operating (continuous) load is less than 128 amperes.
4. All loadings indicated are amperes.

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CHAPTER 9

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9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The fuel pool cooling and cleanup system has been designed to comply with the objectives set forth in Regulatory Guides 1.13, Revision 1 and 1.26 Revision 3 to the extent specified in the following subsections. The system and equipment are designed to the classifications given in Table 3.2-1.

The system is designed to remove the decay heat released from the spent fuel elements and maintain a specified fuel pool water temperature, water clarity, and water level by accomplishing the following:

a. Minimizing corrosion product buildup and controlling fuel pool water clarity so that fuel assemblies can be efficiently handled under water.

b. Minimizing fission product concentration in the fuel pool water thereby minimizing the release of fission products from the pool to the reactor building environment.

c. Monitoring surge tank water level to thereby maintain a pool water level above the fuel sufficient to provide shielding for normal building occupancy and to control make-up flow rate from the condensate transfer system.

d. Maintaining the fuel pool water temperature below 125°F under normal operating conditions. The maximum heat load in the fuel pool under normal operating conditions occurs at the end of the 12th refueling cycle at which time there are 2068 fuel assemblies in the high density fuel racks. The estimated refueling data is given in Table 9.1-3.

9.1.3.2 System Description

The fuel pool cooling and cleanup system flow diagram is shown on Figure 9.1-4. System performance data are summarized in Table 9.1-1. Major components of the system are summarized in Table 9.1-2. The system is designed to dissipate the fuel pool heat load during equilibrium or non-equilibrim fuel cycle conditions.

If required, heat removal capacity is available for the full core removal load during either of these periods, in addition to the spent fuel load already stored. The system design heat load is based on the data given in Table 9.1-3.

The system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the reactor building closed cooling water system. Water purity and clarity in the storage pool, reactor well, and dryer-separator pit are maintained by filtering and demineralizing the pool water through a filter demineralizer. In addition to fuel pool water demineralization, the system will be used on occasion to demineralize suppression pool water.

The pool cooling and cleanup system consists of two circulating pumps, two heat exchangers, two filter demineralizers, two skimmer surge tanks, and the required piping, valves, and instrumentation. The pumps circulate the pool water in a closed loop, taking suction from the surge tanks, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well. The water flows from the pool surface through scuppers and skimmer weirs to the surge tanks. Make-up water for the system is transferred from the condensate storage tank to a skimmer surge tank to make up evaporative losses. The fuel pool pumps and heat exchangers are located in the reactor building beneath the fuel pool.

Fuel pool water is continually recirculated except when draining the reactor well and dryer-separator pit. The operating temperature of 125°F is permitted to rise to 150°F when the circulating flow is interrupted for draining the reactor well and dryer-separator pit, or when larger than normal batches of fuel are stored. The fuel pool cooling and cleanup system is interconnected with the residual heat removal system to supplement the pool cooling during refueling in the event that a larger than normal batch of fuel is stored.

To establish a circulating pattern of flow in the reactor well and storage pool, the diffusers and skimmer drains are placed to sweep particles dislodged during refueling operations away from the work area and out of the pool.

Fuel pool water clarity and purity is maintained by a combination of filtering and ion exchange. The filter demineralizer maintains a total heavy element content (Fe, Cu, Hg, Ni, etc.) of 0.1 ppm or less with a pH range of 6.0 to 7.5.

Particulate material is removed from the water by the pressure precoat filter demineralizer units. The finely divided disposable filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted. The spent filter medium is backwashed to the waste sludge phase separator tank for processing in the solid radwaste handling system. New filter medium is mixed in a precoat tank and is transferred as a slurry by a precoat pump where the solids deposit on the filter elements. The holding pump connected to each filter demineralizer maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation. A strainer is provided in the effluent stream of the filter demineralizers to limit the migration of the filter material.

The two filter demineralizer units are located separately in shielded cells in the radwaste building. Sufficient clearance is provided in the cells to permit removal of the filter elements from the vessels. Each cell contains only the filter demineralizer and its associated piping. All valves are located on the outside of one shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls.

Instrumentation is provided for both automatic and remote manual operation. Indication is provided in the control room and pump room. Surge tank high and low water level switches are provided. A local level indicator is provided to monitor reactor well water level. Control of flow to or from the reactor well can be accomplished during refueling. A fuel pool high/low water level switch operates a local indicator light and sounds an alarm in the control room whenever the level is either too high or too low. The trip point is adjustable over the range of the skimmer weir adjustment.

The pumps are controlled from either the pump room or the vicinity of the fuel pool filters. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarm annunciates in the control room and in the pump room. The controls for the remote controlled fuel pool discharge valves are located on a rack in the pump room. The open or closed condition of each of these valves is indicated by a light in the pump room.

The flow rate through the filter demineralizers is indicated by a flow indicator on the pump room panel.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the operating floor instrument racks and is alarmed in the control room.

The filter demineralizers are controlled from a local panel. Differential pressure and conductivity instrumentation is provided for each unit to indicate when backwash is required. Suitable alarms, differential pressure indicators, and flow indicators are provided to monitor the condition of the filter demineralizers.

9.1.3.3 Safety Evaluation

The maximum possible heat load will be the decay heat of one full core load of the fuel due to an emergency dump into the pool plus the remaining decay heat of previously discharged batches of fuel. The residual heat removal system (RHR) can be operated in parallel with the fuel pool cooling and cleanup system during this condition when the pool has a greater than normal load and when its temperature exceeds 125^{o}F. The RHR system can be used in parallel with the fuel pool cooling system to remove abnormal heat loads, as well as during the normal refueling mode. The RHR system will not be initiated unless the reactor is in a cold shutdown condition. The operator must insert spool pieces in supply and discharge piping and open normally closed valves to permit the use of this system for supplementary cooling.

The fuel pool heat exchangers are cooled by the reactor building closed cooling water system to prevent contamination outside the reactor building in the event of a fuel pool heat exchanger tube failure. The system can maintain the fuel pool water temperature below 125^{o}F when removing the nominal heat load from the pool with the reactor building closed cooling water temperature at its maximum of 95^{o}F. The fuel pool water temperature is permitted to rise to approximately 150^{o}F while the system water flow is diverted from the pool to drain the reactor well and dryer-separator pit, or when larger than normal batches of spent fuel are stored in the pool.

There are no connections to the fuel storage pool which could allow the fuel pool to be drained below the pool gate between the reactor well and the fuel pool. Two diffusers are placed in both the reactor well and the fuel pool to distribute the return water as efficiently and with as little turbulance as possible. Diffusers are placed to minimize stratification of

either temperature or contamination. A check valve is connected to each pipe outside the pool to prevent the pool water from being siphoned out of the pool and uncovering the spent fuel. Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.

A make-up water valve controlled by tank level switches supplies condensate from the condensate transfer system to the pool to replace evaporative and leakage losses. The backup source of make-up water is from the Seismic Category I, Safety Class 3 standby service water system. This connection supplies enough water to prevent the uncovering of the spent fuel. By use of the standby service water as make-up, the fuel pool will be cooled by evaporation of the pool water.

Each filter demineralizer is capable of continual performance, operating at a maximum fuel pool water flow rate of 1000 gpm and will maintain water conditions as specified in 9.1.3.2.

The following components of the fuel pool cooling and cleanup system (FPC) are designed to ASME Section III, Class 3: fuel pool cooling pumps, filter demineralizers, pumps, valves, and piping, FPC piping, and fuel pool heat exchanger. The system heat exchangers are also designed to the standards of the Tubular Exchangers Manufacturers Association, Class R. Piping in the reactor building is controlled and supported to Seismic Category I requirements. The water lines between the fuel pool and RHR systems are designed to ASME Section III, Class 3, Seismic Category I requirements . The FPC pumps are not designed to Seismic Category I requirements. Condensate piping in the reactor building is controlled and supported to Seismic Category I requirements.

A radiological evaluation of the cleanup system is presented in Chapter 12.

From the foregoing analysis, it is concluded that the fuel pool cooling and cleanup system meets its design basis and satisfies the requirements of Regulatory Guide 1.13, Revision 1 with exceptions as noted in this section.

9.1.3.4 Testing and Inspection Requirements

No special tests are required because at least one pump, heat exchanger, and filter demineralizer are continuously in operation while fuel is stored in the pool. Duplicate components are operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation, and trouble alarms are adequate to verify system operability.

TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM EQUIPMENT DATA

Fuel Pool Heat Exchangers

Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/heat exchanger	4.0×10^6
Cooling Water Flow, gpm/heat exchanger	575
Code and Standards	ASME/III-Class 3 and TEMA-Class R
Seismic Category	II

Fuel Pool Circulation Pumps

Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	575
Head, Ft of H_2O	160
Motor Size hp	40
Seismic Category	II
Code	ASME/III-Class 3

Fuel Pool Filter Demineralizer

Number	2
Design Flow Rate, gpm	1000 Maximum, 575 Normal
Design Pressure, psig	150
Design Temperature, ^oF	150
Material	CS-Plastic Lined
Code	ASME/III-Class 3
Seismic Class	II

Piping and Valves

Design pressure, psig	150/300
Design Temperature, ^oF	220
Material	CS/SS
Code	ASME/III-Class 3

TABLE 9.1-3

ESTIMATED REFUELING CYCLE DATA***

Cycle* Number	Number of Fuel Assemblies Discharged	Equivalent Irradiation Time (Days)	Decay Time (Days)
1	204	333.718	4035
2	200	662.325	3670
3	168	914.470	3305
4	152	998.287	2940
5	160	1043.492	2575
6	160	1095.003	2210
7	160	1139.089	1845
8	160	1143.125	1480
9	176	1275.495	1115
10**	176	1271.148	750
11**	176	1257.620	385
12**	176	985.186	20
Total	2068		

* The word "cycle" is used here to describe the spent fuel leaving the reactor at the end of the irradiation cycle having the same number.

** Uranium fuel with plutonium added.

*** Normal ("average") conditions except last four years are low stream flow (high plant capacity factor).

Elevation differentials between the low-low level setting and the HPCS pump impeller (Elev. 420'-4-1/2") and the RCIC pump impeller (Elev. 427'-3") provide a suction head of at least 20 feet during HPCS/RCIC operation. (Tank bottom elevation is at 443 ft.)

Thermostatically controlled tank heaters are provided to maintain water temperature in the tanks at or above a nominal 40°F at all times. All above ground piping that contains water is heat traced to prevent freezing.

System logic diagrams are given in Chapter 7.

9.2.7 STANDBY SERVICE WATER SYSTEM

9.2.7.1 Design Bases

a. The standby service water system (SW) is designed to remove heat from plant systems which are required for a safe reactor shutdown following a LOCA.

b. The system is designed to remove reactor decay heat from the residual heat removal system during normal plant shutdown.

c. The system is designed to perform its required cooling water function following a LOCA, assuming a single active failure.

d. The system is designed to provide a means of flooding the vessel and containment, if required during the post-LOCA period.

e. The system is designed to provide makeup source of water for ensuring fuel pool evaporative cooling following a LOCA in conjunction with a design basis earthquake.

f. The system is designed to Seismic Category I and ASME Code, Section III, Class 3 requirements with the exception of that portion to and from the plant cooling towers, which is designed to ANSI B31.1 and Seismic Category II requirements.

9.2.7.2 System Description

The standby service water system includes vertical service water pumps located adjacent to the two spray ponds in two

separate pumphouses designed to Seismic Class I criteria. The pumps discharge to three independent piping systems which serve emergency core cooling system equipment, auxiliary plant equipment and reactor shutdown cooling equipment. (See Figure 9.2-10 through 9.2-13).

During normal shutdown cooling, control of biological growth is provided by the circulating water system. (See 10.4.5).

The spray ponds are provided with makeup water by the circulating water system. The makeup water system supplies Columbia River water to the cooling towers or spray pond to replace water lost during normal operation due to evaporation and drift. In addition, the makeup system is designed to replace spray pond water lost during a tornado. To ensure system availability for this mode of operation, the makeup system is designed to withstand a design basis tornado coincident with a loss of offsite power.

The standby service water system piping is carbon steel designed to 300 psig, 150°F, and with a corrosion allowance of 0.080 inches.

Emergency makeup water to the fuel pool can be supplied through normally locked closed valves on 2" lines upstream of both "A" and "B" RHR heat exchangers.

EQUIPMENT DESIGN PARAMETERS

Standby Service Water Pumps (SW-P-1A, SW-P-1B)

Quantity	2
Driver	Motor - AC
Design Capacity	10,500 gpm
Head	500 ft.

HPCS Service Water Pump (HPCS-P-2)

Quantity	1
Driver	Motor - AC
Design Capacity	1,200 gpm
Head	123 ft.

9.2.7.3 Safety Evaluation

The standby service water system provides cooling for plant equipment which is essential to a safe reactor shutdown following a design basis loss-of-coolant accident. The entire system is adequately protected to withstand the following adverse environmental occurrences:

a. Design Basis Earthquake

b. Design Basis Wind Loads

c. Tornado

Redundant trains of the standby service water system are separated and protected to the extent necessary to ensure that sufficient equipment remains in operation to permit safe shutdown of the unit in the event of any of the following events:

a. Flooding or steam release from equipment failure such as pipe or tank rupture.

b. Pipe whip and jet forces resulting from pipe rupture.

c. Missiles which may result from equipment failure.

d. Fire.

The standby service water system is designed to withstand a single active failure without losing its capability to participate in the safe shutdown of the reactor following an accident.

System failure mode and effects analyses of passive and active components of the service water system is presented in Table 9.2-6. Any of the assumed failures of the service water system is detected in the main control room by indications and/or alarms from the various system instruments.

The standby service water is routed through the tube side of the RHR heat exchanger and through the shellside of the RHR pump seal coolers. The RHR heat exchanger and the pump seal cooler are the only potential sources of radioactive leakage to the standby service water system.

A manually set and locked control valve is provided in the return line of each of the two standby service water sub-

systems to maintain the pressure in the system at a level such that tube leakage in the RHR heat exchanger or pump seal cooler is from the service water system into the RHR system after the reactor is depressurized following shutdown cooling or after a LOCA. Liquid radiation monitors are provided in each of the two RHR heat exchangers outlet lines of the service water system to detect radioactivity resulting from a tube leak in one of the RHR heat exchangers or leakage from the pump seal coolers. This condition can occur only when the RHR system is operated in a mode in which the service water system pressure is lower than the RHR system pressure. Upon detection of radioactive leakage in one of the subsystems, that subsystem is isolated by operator action in the main control room and the cooling requirements are met by the redundant train. Consequently, radioactivity released to the spray ponds and/or cooling tower basins is minimized.

An intertie with the RHR system is provided from the "B" standby service system supply header which contains two remote manually operated isolation valves. These valves can be opened from the main control room in the event primary containment flooding is required following a loss-of-coolant accident.

The standby service water pumps are provided with double valved, normally closed bypass connections for transferring water from one pond to another. Should the service water pump be unavailable for transfer purposes, an atmospheric syphon is available for backup service.

Temperature controlled and/or manually operated throttle valves are located on the return side of all standby service water system coolers and heat exchangers. The system is balanced for optimum operation and the throttle valves left in that position. The RHR heat exchanger service water outlet valve is interlocked to open on starting of a standby service water pump to prevent excess flow conditions in other portions of the system.

Redundant spray pond low level switches in each spray pond pumphouse automatically close the redundant isolation valves on the SSW return line to the cooling towers and open the isolation valve to the spray pond. The ECCS start signals also automatically perform this function to ensure that the spray ponds always maintain sufficient inventory to meet the system requirements delineated in 9.2.5.

During plant operation, the standby service water pumps are not operating. If a LOCA occurs, the diesel generators and respective standby service water pumps start automatically. Consequently, no operator action is required following a LOCA to start the standby service water system and put the system into its LOCA operating mode.

9.2.7.4 Testing and Inspection Requirements

The standby service water system is designed to permit periodic inspection of all active system components to ensure the integrity and capability of the system. For inservice inspection see 6.6.

The system is designed to permit periodic pressure and functional testing to ensure the following:

a. The structural and leaktight integrity of its components.

b. The operability and the performance of the active components of the system.

c. The operability of the system as a whole.

The system is performance tested during plant operation using the operational sequence that brings the system into operation for reactor shutdown and for a LOCA. Motor operated isolation valves are tested to ensure that they are capable of opening and closing by operating manual switches in the main control room.

The system is operated and tested initially with regard to flow paths, flow capacity and mechanical operability. Startup is described in Chapters 14 and 16.

9.2.7.5 Instrumentation Requirements

Each of the standby service water discharge lines from the RHR heat exchangers contains radiation monitors (RE-N004, RE-N005) to detect any radioactivity resulting from a tube leak in the heat exchanger.

Flow indicators and/or switches are provided for each service component to indicate low flow. Temperature indicators, temperature switches, pressure indicators and/or pressure switches are located in each system to determine pump, individual cooling coil, cooler or heat exchanger performance. For detailed system logic diagrams and electrical elementaries see Chapters 7 and 1 respectively.

In order to avoid excessive system surge pressures, the standby service water pumps are started only if the associated pump discharge valve is closed. The pumps start automatically when the associated diesel generator is started. The pump discharge valve automatically opens when the associated pump is running and the pump discharge pressure is greater than 50 psig. The system instrumentation is shown on system flow diagram, Figure 9.2-10.

TABLE 9.2-1

ULTIMATE HEAT SINK SPRAY COOLING
POND DESIGN DATA

Pond configuration	Square (250' x 250')
Surface area (two ponds)	125,000 sq. ft.
Normal water elevation (above MSL)	434'-0"
Maximum water elevation (above MSL)	434'-6"
Pond bottom elevation (above MSL)	420'-0"
Pump sump bottom elevation (above MSL)	408'-3"
Freeboard above normal water level	0'-6"
Sedimentation allowance	0'-6"
Normal pond capacity (two ponds)	12,500,000 gallons

TABLE 9.2-4 (Continued)

DIURNAL VARIATION IN METEOROLOGICAL DATA (FOR DAY 1 THRU 30 USED TO ANALYZE MASS LOSS FOLLOWING LOCA)

Hour	Dry Bulb (^{o}F)	Dew Point (^{o}F)	Wet Bulb (^{o}F)	Wind* Speed (mph)	Solar Radiation ($\frac{BTU}{hr}$)
Noon	96.40	42.50	64.70	10.30	290.81
1:00 p.m.	98.00	43.50	65.40	10.30	282.71
2:00	98.50	43.50	65.60	10.30	261.30
3:00	98.00	43.50	65.40	10.30	226.27
4:00	96.40	42.50	64.70	10.30	180.98
5:00	93.90	42.00	63.70	10.30	127.56
6:00	90.70	42.00	62.30	10.30	70.89
7:00	86.90	40.50	60.70	10.30	16.86
8:00	82.90	40.00	59.00	10.30	0.00
9:00	78.90	40.00	57.30	10.30	0.00
10:00	75.10	39.00	55.70	10.30	0.00
11:00	71.90	39.00	54.30	10.30	0.00
Midnight	69.40	39.00	53.30	10.30	0.00
1:00 p.m.	67.80	39.00	52.60	10.30	0.00
2:00	67.30	39.00	52.40	10.30	0.00
3:00	67.80	39.00	52.60	10.30	0.00
4:00	69.40	39.00	53.30	10.30	0.00
5:00	71.90	39.50	54.30	10.30	16.86
6:00	75.10	39.00	55.70	10.30	70.89
7:00	78.90	40.00	57.30	10.30	127.56
8:00	82.90	40.00	59.00	10.30	180.98
9:00	86.90	40.70	60.70	10.30	226.27
10:00	90.70	42.00	62.30	10.30	261.30
11:00	93.90	42.20	63.70	10.30	282.71

Data based upon average values for the period 2 July - 1 August 1960.

* Wind speed is the highest daily average wind speed for the period.

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TABLE 9.2-5

EQUIPMENT REQUIRING STANDBY SERVICE WATER TO ENSURE PLANT SHUTDOWN

Equipment Cooled	Required Flow-gpm(1)	Design Heat Load (Btu/hr)	Calculated Heat Load (Btu/hr)
Division I			
1. Standby Service Water Pumphouse "A" Cooler	80	404,000	380,600
2. Diesel Generator "A"	1650 (2)	15,600,000	11,692,427
3. Diesel Generator Building "A" Coolers	144	716,000	716,000
4. LPCS Pump Motor Bearings	4 (3)	-	∿0
5. LPCS Pump Room Cooler	56	280,000	270,860
6. RHR "A" Pump Seals	12 (2)	-	∿0
7. RHR "A" Room Cooler	33	165,000	149,650
8. D.C. Motor Control Center Room Cooler	20	84,200	40,533
9. Motor Control Center Room Cooler	15	71,280	43,130
10. Control Room Cooler	120	285,000	256,500
11. Cable Spreading Room Cooler	40	160,000	74,600
12. Switchgear Room Cooler	60	370,000	327,100
13. Hydrogen Recombiner "A" MCC Room Cooler	11	52,500	36,174
14. Hydrogen Recombiner "A" Aftercooler	50	-	250,000
15. Hydrogen Recombiner "A" Scrubber	10	-	50,000
16. RHR "A" Heat Exchanger	7400 (2)	(4)	Variable
17. Analyzer Room Cooler	10	42,500	23,571
TOTAL	9715		

1) Based on 85°F Standby Service Water Supply unless otherwise noted.
2) Design based on 95°F Standby Service Water Supply
3) Design based on 90°F Standby Service Water Supply
4) See Table 6.2-2 for design parameters

TABLE 9.2-5 (Continued)

Equipment Cooled	Required Flow-gpm(1)	Design Heat Load (Btu/hr)	Calculated Heat Load (Btu/hr)
Division II			
1. Standby Service Water Pumphouse "B" Cooler	80	404,000	358,100
2. Diesel Generator "B"	1650 (2)	15,600,000	11,692,427
3. Diesel Generator Building "B" Coolers	144	716,000	716,000
4. Diesel Generator Area Cable Cooler (Corridor)	40	149,000	109,680
5. RHR "B" Pump Seals	12 (2)	-	~0
6. RHR "C" Pump Seals	12 (2)	-	~0
7. RHR "B" Room Cooler	33	165,000	145,650
8. RHR "C" Room Cooler	33	165,000	160,530
9. RCIC Pump Room Cooler	12	60,000	37,270
10. Motor Control Center Room Cooler	15	71,280	43,130
11. Control Room Cooler	120	285,000	256,500
12. Cable Spreading Room Cooler	40	160,000	74,600
13. Switchgear Room Cooler	60	320,000	305,400
14. Hydrogen Recombiner "B" Aftercooler	50	-	250,000
15. Hydrogen Recombiner "B" Scrubber	10	-	50,000
16. Hydrogen Recombiner "B" MCC Room Cooler	11	52,500	36,174
17. RHR "B" Heat Exchanger	7400 (2)	(3)	Variable
18. Analyzer Room Cooler	10	42,500	23,571
TOTAL	9732		

1) Based on 85°F Standby Service Water Supply unless otherwise noted.
2) Design based on 95°F Standby Service Water Supply
3) See Table 6.2-2 for design parameters

TABLE 9.2-5 (Continued)

Equipment Cooled	Required Flow-gpm(1)	Design Heat Load (Btu/hr)	Calculated Heat Load (Btu/hr)
Division III			
1. HPCS Diesel Generator	910 (2)	8,872,000	7,401,000
2. HPCS Diesel Building Coolers	144	716,000	716,000
3. HPCS Pump Room Cooler	50	500,000	473,580
TOTAL	1104		

1) Based on 85°F Standby Service Water Supply unless otherwise noted.
2) Design based on 95°F Standby Service Water Supply

The air compressors take suction from the room through filter silencers which are 95% efficient in filtration of particles as fine as three microns. They discharge to the air receivers through intercoolers and aftercoolers for cooling and moisture separation. Moisture condensed in the coolers is discharged through float traps. Cooling water is supplied to the coolers by the plant service water system. Service and control air is directed from the air receivers to the system headers.

Service air is distributed from the header to quick-disconnect hose connections where it is used for pneumatic service equipment and maintenance throughout the plant. Service air is also distributed for plant services such as demineralizer resin mixing and filter and demineralizer backwashing (see Figure 9.3-1). In order to ensure the availability of the control air system for turbine and reactor plant operation, an air operated isolation valve is provided to isolate the service air system when the pressure in the service air header drops to 75 psig.

Control air for station instrumentation and controls is directed from the system header and is processed through one of two 100% capacity prefilters, an air dryer, and one of two 100% after filters prior to distribution for use. The filters are of the removable cartridge type and are arranged in parallel pairs to allow for replacement of a prefilter or an afterfilter without interruption of air flow. The twin tower air dryer contains a regenerable desiccant and is operated so that one section is in use while the other is being regenerated (four hours in service, four hours reactivation).

Table 9.3-1 presents the major characteristics of the air compressors, receivers, and the air dryer for the control and service air system.

9.3.1.2.2 Containment Instrument Air System

The containment instrument air system is shown in Figure 9.3-2, and the major equipment characteristics are presented in Table 9.3-1. The location of this system in the Seismic Category I reactor building is shown in Figures 1.2-3 and 1.2-5.

This system consists of two 100% capacity air compressors, associated coolers, a twin tower air dryer, filters, an air receiver, valves, and piping of a leak tight design. In addition, two nitrogen gas bottle banks and associated piping are provided as a backup to the compressor supplied air for seven of the main steam relief valves which perform the ADS function.

The compressors located in the reactor building take suction from the building atmosphere through intake filter-silencers which are 98% efficient in filtration of particles as fine as five microns. The air is then discharged through an aftercooler, a prefilter, a dryer, an afterfilter and air receiver to deliver dry, clean, pressurized air to the pneumatic control systems of the following valves inside the primary containment vessel:

a. Four main steam isolation valves and their accumulators,

b. Eighteen main steam safety/relief valves and their accumulators.

The two independent nitrogen bottle bank subsystems are provided to deliver pressurized nitrogen to seven of the safety/relief valves and accumulators. These seven valves perform the ADS function, if required, during postulated LOCA conditions. These nitrogen banks ensure a 30-day supply of nitrogen for the ADS function during isolation of the compressor loop. One bank of 15 bottles supplies nitrogen for three main steam safety/relief valves and accumulators, while the other bank of 19 bottles supplies four main steam safety/relief valves and accumulators (see Figure 9.3-2).

The nitrogen bottles are located in the railroad lock of the reactor building to facilitate access. Under normal operating conditions, the controlled leakage boundary of the Reactor Building is maintained above the railroad lock so access is available to the bottles for recharging if required. The bottles are standard, commercially available units pressurized to 2490 psig. Each bottle has a capacity of 257 SCF. However, the bottles are mounted in accordance with Seismic Category I, Quality Class I requirements. The required quantity of bottles for each bank was conservatively based on providing a 30-day supply to the ADS valves to satisfy the long term post-LOCA demand based on the following:

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, the boron can be removed from the ractor coolant system by flushing, followed by operating the reactor cleanup system. There is practically no effect on reactor operations when the boron concentration has been reduced below approximately 50 ppm.

The concentrationof the sodium pentaborate in the solution tank is determined periodically by chemical analysis.

Electrical supplies and relief valves are also subjected to periodic testing (See Chapter 16).

The SLC system preoperational test is described in 14.2.12.1.5.

9.3.5.5 Instrumentation Requirements

The instrumentatin and control system for the SLC system is designed to allow the injection of neutron absorber solution into the reactor and to maintain the neutron absorber solution well above the saturation temperature. A further discussion of the SLC system instrumentation may be found in 7.4.1.2.

TABLE 9.3-1 Page 1 of 2

EQUIPMENT CHARACTERISTICS

CONTROL AND SERVICE AIR SYSTEMS

a.	Air Compressors	
	1. Quantity	3
	2. Rated Output (each)	450 SCFM
	3. Rated Discharge Pressure	115 PSIG
b.	Air Receivers	
	1. Quantity	3
	2. Volume (each)	96 Ft^3
c.	Air Dryer	
	1. Quantity	1
	2. Type	Twin Tower - Regenerative
	3. Media: Active	Activated Alumina Absorbent
	Buffer	Silica Gel
	4. Rated Inlet Drying Capacity (each)	750 SCFM
	5. Dew Point at Outlet	$-40^{o}F$
	6. Cycle of Operation	8 Hrs.

CONTAINMENT INSTRUMENT AIR SYSTEM

a.	Air Compressors	
	1. Quantity	2
	2. Rated Output (Min. each)	50 SCFM
	3. Rated Discharge Pressure (Min.)	160 PSIG
b.	Air Receiver	
	1. Quantity	1
	2. Volume	34 Ft^3

Room	Maximum Allowable Ambient Temp. (°F)
HPCS Pump Room	148°
Div. 1 LPCS Pump Room	148°
Div. 2 RCIC Pump Room	148°
Div. 1 RHR Pump Room	148°
Div. 2 RHR Pump Rooms (2 Pump Rooms)	148°
Div. 1 MCC Room (EL. 522)	104°
Div. 2 MCC Room (EL. 522)	104°
Div. 1 D.C. MCC Room (EL. 471)	104°
Div. 1 H_2 Recombiner MCC Room (EL. 572)	104°
Div. 2 H_2 Recombiner MCC Room (EL. 572)	104°
Div. 1 Reactor Building Analyzer Room 1A	104°
Div. 2 Reactor Building Analyzer Room 1B	104°

The electrical equipment rooms are isolated from the reactor building heating and ventilating system upon a signal of building isolation.

All components of the reactor building emergency cooling system are designed as engineered safety features and are powered from the same diesel generator bus as the equipment being served. Since each separate cooling system services redundant emergency equipment systems, a failure of one cooling system will not effect the operational function of the other cooling systems or the safe shutdown of the reactor. The means of protecting the system vents and louvers from missiles is discussed in 3.5.

All ductwork connected to the fan coil units in this system is designed to Seismic Category I requirements. The system fans are constructed and rated in accordance with the applicable AMCA standards. The water cooling coils are designed and code stamped in accordance with the requirements of the ASME Code, Section III Class 3.

9.4.9.2 System Description

The reactor building emergency cooling system is depicted in Figure 9.4-2. Each of the thirteen rooms housing critical equipment is provided with an individual air handling unit which is fully enclosed within the room. Each air handling unit is comprised of a direct drive centrifugal fan and a water cooling coil in a sheet metal housing. Water is supplied to the water coils by the standby service water system (see 9.2.7). During normal operation, all thirteen air handling units are in standby. The units servicing the pump rooms start upon actuation of their associated pumps. The units servicing the MCC equipment rooms and the analyzer rooms start automatically upon any signal which isolates the reactor building.

All units recirculate the air within the room they serve, removing the heat generated in the room via the water coil, to maintain temperatures below the design limits listed in 9.4.9.1.

9.4.9.3 Safety Evaluation

Each of the thirteen emergency equipment rooms in the reactor building is provided with a separate, independent cooling system, all components of which are located within the room serviced. Each cooling system is powered from a Class 1E bus of the same division as the equipment it serves and is designed to withstand the effects of a safe shutdown earthquake. A failure of one cooling system will not affect the operational function of any other system or the safe shutdown of the reactor.

The air handling units serving the six pump rooms are interlocked electrically with the pumps they serve in such a manner that they start when the pump is started. The seven air handling units serving the critical MCC rooms and the analyzer rooms are started by any of the following three isolation signals:

a. Reactor vessel low water level

b. High drywell pressure

c. High radiation level in reactor building exhaust ventilation system

9.4.9.4 Testing and Inspection Requirements

All components of the reactor building emergency cooling system are normally in standby and are accessible for out-of-service inspection. All system ductwork and components are subjected to leak tests and all piping systems are subjected to hydrostatic tests during manufacture and erection.

The cooling system is subjected to pre-operational testing at design conditions and performance shall be verified periodically by testing during unit operation.

9.4.9.5 Instrumentation Requirements

The six air handling units serving the pump rooms are controlled identically. Each is electrically interlocked with the pump it serves to operate when the pump operates. Running lights for the fans are located in the main control room on the same panels as the pump switches. The standby service water system supplies the air handling unit water coil when the pump is started. A local manual switch is provided in each pump room for testing the air handling unit fan.

The controls for each of the seven cooling systems serving the critical MCC and analyzer rooms are also identical. An ON-AUTO-OFF switch (spring back from OFF to AUTO) is provided for each unit in the main control room. Normally all switches are in the AUTO position and the units are in standby. In the AUTO mode, any of the three isolation signals listed in 9.4.9.3 will cause the following operations, via electric interlocks:

a. Air handling unit fans start.

b. Standby service water system is energized when the associated emergency diesel generators start.

c. Solenoid valves associated with the air operated dampers in the reactor building ventilation system supply air ducts to the MCC and analyzer rooms are de-energized, thus isolating these rooms from the balance of the reactor building.

Temperature indicators for each of the thirteen equipment rooms are provided in the main control room. Separate temperature switches in each room will annunciate an alarm in the main control room in the event that temperatures exceed the design limit.

9.4.10 STANDBY SERVICE WATER PUMP HOUSE VENTILATION SYSTEMS

9.4.10.1 Design Bases

The standby service water pumphouse heating and ventilating systems are designed to remove the heat generated by operation of the standby service water pumps and the HPCS service water pump and to limit the temperature in the two pump houses to 114°F. The ventilation systems are designed as engineered safety feature systems and are powered from Class 1E bus of the same division as the pumps being served. Since each heating and ventilation system associated with the redundant standby service water pumps is separate and independent, a failure in one system will not affect the operational function of the other system. The means of protecting the system vents and louvers from missiles is discussed in 3.5.

Heating of the pump houses is provided by electric unit heaters (four in each building) sized to heat the building spaces to approximately 40°F during outside winter temperature of -27°F regardless of the pump operational mode. The unit heaters are powered from the emergency diesel-generator buses, but are classified Seismic Category II. The remaining components, ductwork, and piping in the standby service water pump house ventilation systems are classified Seismic Category I.

The fan coil units and the supply fans are constructed and rated in accordance with the applicable AMCA standards. The water cooling coils are designed and code stamped in accordance with the requirements of ASME Code, Section III Class 3.

9.4.10.2 System Description

The standby service water pumps are located in pump houses adjacent to the emergency spray ponds. The loop "A" standby service water pump and the HPCS service water pump share one pump house and the loop "B" standby service water pump is in a second pump house. The heating and ventilating systems serving the two pump houses are depicted in Figure 9.4-7. Each system consists of a 17,000 cfm fan coil unit composed of a sheet metal cabinet containing a direct-drive centrifugal fan and a water cooling coil, and a separate 5,000 cfm centrifugal supply fan with inlet mixing dampers.

Each fan coil unit is interlocked electrically with the associated standby service water pump it serves in such a manner that the unit fan starts and recirculates room air over the water cooling coil when the pump starts. Water is supplied to the unit coil from the main supply header of the standby

TABLE 9.4-2

REACTOR BUILDING AND PRIMARY CONTAINMENT AREAS MAJOR COMPONENTS OF HVAC SYSTEMS

a. Heating and Ventilating Unit (Evaporative Air Washer)

1.	Tag No.	ROA-HV-1
2.	No. of Units	1
3.	Air Flow (SCFM)	97,350
4.	Steam Flow (lbs/hr)	6,800
5.	Supply Air Temperature Summer Winter	 72°F DB (Evap. Cooling) 50°F
6.	Fan Type	Vaneaxial V-belt Drive (200 H.P. Motor)
7.	No. of Fans	2 (one standby)
8.	Fan Total Pressure (inches, w.g.)	9.3

b. Reactor Building Exhaust Fans

1.	Tag No.	REA-FN-1A & 1B
2.	No. of Fans	2 (one standby)
3.	Fan Type	Vaneaxial
4.	Drive	Direct (200 H.P. Motor)
5.	Capacity	95,000 - 105,000
6.	Total (SCFM) Pressure (inches, w.g.)	9.43 @ 105,000 SCFM
7.	Motor Type	TEAO with Class "RH" insulation
8.	Seismic Category	II

TABLE 9.4-2 (Continued)

c. Reactor Building Emergency Fan Coil Units

1.	Tag No.	RRA-FC-1 RRA-FC-2 RRA-FC-3	RRA-FC-4	RRA-FC-5	RRA-FC-6	RRA-FC-10 RRA-FC-11
2.	No. of Units	3	1	1	1	2
3.	Air Flow per Unit (ACFM)	5208	15,625	9375	3125	5730
4.	Sensible Cooling Capacity (Btuh) per unit	165,000	500,000	280,000	60,000	71,280
5.	Total Static Pressure (inches,w.g.)	1.34	1.64	1.46	1.53	0.5
6.	Area Served	RHR Pump Rooms	HPCS Pump Room	LPCS Pump Room	RCIC Pump Room	MCC Rooms
7.	Water Supply Service	Standby Service Water	Standby Service Water	Standby Service Water	Standby Service Water	Standby Service Water
8.	Seismic Category	I	I	I	I	I

9.5.4 DIESEL GENERATOR FUEL OIL STORAGE AND TRANSFER SYSTEM

9.5.4.1 Design Bases

a. The fuel oil storage and transfer system for the standby diesel generators is designed to perform its operational function during emergency conditions assuming any single active or passive failure of one of its components.

b. The onsite storage capacity of each subsystem provides for continuous operation of each diesel generator for at least seven days while satisfying post-LOCA maximum load demands.

c. The design of the system conforms to IEEE Standards 308 and 387 and ANSI Standard N195. The equipment within the system conforms to the applicable codes and standards of ASME, ASTM, ANSI, DEMA, IEEE, API, and NFPA.

d. The system piping is designed to ASME Section III, Class 3 and Seismic Category I requirements. (See Table 9.5-6 for equipment design codes). Except for the diesel oil storage tanks, all portions of the system, including the fuel oil day tanks; are protected from tornado missiles by enclosure in Seismic Category I structures. The diesel oil storage tanks are buried for protection and to maximize containment of postulated oil spills. The system is not subject to flooding since the site is not subject to flooding.

9.5.4.2 System Description

The fuel oil storage and transfer system consists of separate, independent diesel oil supply subsystems serving each of the two tandem diesel engine generators (1A and 1B) and the HPCS diesel engine generator. Each of these subsystems consists of a 100% capacity fuel oil storage tank, a transfer pump, one day tank, interconnecting piping and valves, and associated instruments and controls. The fuel oil transfer piping of diesel generators 1A and 1B is arranged such that either transfer pump may transfer diesel oil from its storage tank to either day tank, if required. The system diagram is shown on Figure 9.5-4.

In each supply subsystem, a transfer pump powered from a UPS bus takes suction from the diesel oil storage tank and discharges to an associated diesel generator fuel oil day tank to maintain the fuel oil level within the day tank. The transfer pump is sized to provide a flow of 4.4 times the maximum engine consumption rate and is automatically controlled by level switches activated by day tank fuel level. The capacity of each fuel oil storage tank is sufficient to provide 7 days of operation for the diesel generator being served.

Each transfer pump is connected to a day tank. There is a pipe interconnecting the transfer lines. By shutting off the isolation valves in the cross interconnecting line, the fuel oil can be pumped from storage tank "A" and from storage tank "B" to day tank "B". By shutting off the isolation valves at day tank "A" and at transfer pump "B" and opening of the isolation valves is the cross connection line the fuel oil can be pumped from storage tank "A" to day tank "B". By the same logic the fuel oil can be pumped from storage tank "B" to day tank "A". If a rupture occurs in the transfer line between one storage tank and its associated day tank, the interconnecting cross line will be isolated by shutting off the isolation valves in that line. This will assure adequate fuel oil supply to the other day tank. If a rupture occurs in the interconnecting cross line, this line will be isolated and thereby the fuel oil supply will not be interrupted between any storage tank and its associated day tank.

The volume of the day tanks permits eight and one-half hours of engine operation of the associated diesel generator without resupply to the day tank. This arrangement provides three hours of operation before the transfer pumps starts, two hours of operation between a start signal to the transfer pump and day tank low level alarm in the event the transfer pump does not start, and three and one-half more hours of operation after low level alarms are actuated to take required corrective action.

At normal high oil level a switch will shut off the pump. If the fuel oil level reaches two inches above the normal high level, a high high condition is reached and a second level switch is activated. This switch will close the solenoid valve at the day tank inlet, trip the transfer pump and send an alarm signal. Any excess oil will return to the fuel oil storage tank through the one half inch minimum flow line.

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10.4.5.3 Safety Evaluation

The circulating water system is a non-safety related system. Consequently, the circulating water system is not designed to Seismic Category I requirements. Refer to 9.2.5 for a description of the ultimate heat sink which is designed to perform safety-related functions.

The condenser design assures that the pressure on the tube side is always maintained higher than the pressure on the shell side, thus eliminating leakage into the circulating water system should tube failure occur. Consequently, the design of the circulating water system precludes radioactive leakage into the system.

Periodic injection of chlorine is performed for biocide treatment, and sulfuric acid is added for scale-corrosion control within the circulating water system. An analysis of the transportation, handling, storage, and utilization of chlorine is presented in 6.4.

A detailed evaluation was performed to determine the effects of a postulated failure in the circulating water system inside the turbine building. For this analysis a moderate energy crack was postulated to occur in the circulating water system barrier, (e.g., the rubber expansion joints) at the inlet to the main condenser. The inlet side was selected because it yields the severest results.

The entire condenser area is drained by means of sumps (see Figure 9.3-9), each equipped with duplex pumps. Sumps T-2 and T-3, servicing the inlet and outlet of the condenser, each have 50 gpm pumps. Each of these sumps is equipped with a level alarm and is therefore capable of detecting a circulating water system barrier failure. The level alarm will annunciate in the main control room upon reaching high level, providing a means of detecting the postulated failure within 5 minutes.

The crack area for this postulated failure was assumed to be equal to 1/2 the pipe diameter times 1/2 the pipe wall thickness.

$$A = \frac{d}{2} \times \frac{t}{2} \quad \text{(see 3.6.2.1.4.2.b)}$$

The flow exiting from such a crack would be an orifice flow. The head at expansion joint for normal three pump operation at 186,000 gpm each was determined (from system energy gradients) to be 90 feet. The flow for these conditions was calculated to be:

$$Q = 1{,}737 \text{ gpm}$$

The system has different operating pressures for the various modes of pump operation. The piping was designed for an internal pressure of 60 psig, which is well above the design energy gradient.

The motor operated inlet and outlet valves at the condenser are designed and manufactured to close in 60 seconds to avoid excessive pressures caused by fast valve closure. Therefore, rapid valve closure is not a consideration. After closure of the inlet and outlet valves, however, the system will be operating with 2/3 of the condenser capacity. With 3 circulating water pump operation and 2 sections of the condenser in operation, the system flow as determined from the pump operating point diagram will be approximately 450,000 gpm. Comparing the system energy gradients for this mode of operation to that when all three condenser units are in operation, the resultant difference in pressures will be:

> At the inlet side, an increase of approximately 4.3 ft. of head (2 psi) occurs
>
> At the outlet side, a decrease of approximately 5.2 ft of head (2 psi) occurs

Detection of the postulated failure will occur within 5 minutes, as described above, by the annunciation in the control room of the sump high level alarm. It is assumed that there will be a 15 minute time allowance for an operator in the control room to check the circulating water system barriers and close both the inlet and outlet valves of one unit of the condenser as may be required. This closure is accomplished by the activation of a remote manual switch in the control room, and therefore no control circuitry time delays nor coastdown times are involved. Flow will continue, however, after valve closure for about 106 minutes at a decreasing rate, until the remaining water from the condenser is completely discharged.

In the first 5 minutes after a crack, 8,435 gallons of water will spill into the inlet basin. The capacity of each basin and its capability to store excess flow were calculated to be as follows:

a. Inlet basin: 22,500 gallons from El. 436 to El. 441

b. Outlet basin: 27,500 gallons from El. 436 to El. 441

c. Net volume under condenser: 180,500 gallons from El. 433 to El. 441.

The time required to fill the inlet basin, after a postulated crack occurs, is computed to be 13.3 minutes. This includes the 50 gpm outflow from the sump pump. The circulating water leakage flow will continue for 6.7 minutes after filling the inlet basin, until reaching the total estimated shutoff time of 20 minutes. It can be assumed that 10% of this water will flow out over the floor at El. 441, and the remainder, about 10,170 gallons, will flow into the condenser basin area. During this same time period, 4 sump pumps in the condenser basin area will have alternately pumped out 670 gallons, leaving 9500 gallons or 0.42 feet of water in the condenser basin. The rate of rise of water, therefore, is 0.021 ft/min during the first 20 minutes after the postulated crack occurs. Note that on high sump level, both pumps run simultaneously rather than alternately, thus doubling the calculated outflow capacity.

After the valves are closed, the water contained in the condenser unit water box will continue to discharge to the area. The quantity of water remaining is estimated to be 87,000 gallons. The flow will vary with a diminishing head, the head going from about 25 feet to zero feet. Using a 20 ft head and the same orifice flow criteria, the rate of flow will be approximately 819 gpm, discharging the remaining water in about 106 minutes. There will be an outflow from all the sump pumps of 150 gpm, with 10% of the flow from the crack again assumed to flow out over the floor. The water will accumulate in the condenser basin at about 590 gpm. After 106 minutes, the water level in this basin will rise an additional 2.77 feet, or 0.0261 ft/min. The total height of water when the discharge has stopped is therefore 3.19 feet to El. 436.19.

There are no safety-ralated system components that could be affected by the flood elevation established above. Additionally, there are no safety-related electrical systems or system components that could be potentially submerged. In addition, the circulating water piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, spray effects are of no consequence. The pipes exit the rooms below grade in their routing to and from the cooling towers. Also, the floor onto which water would spill in event of a break is grade level. As a result, excess water would accumulate either in drainage basins or leak outside the building.

Discharge operation of water accumulated under the condenser shall be performed in accordance with radioactivity checking requirements for sump discharges.

10.4.5.4 Tests and Inspections

All system components, except the condenser, are accessible during operation and may be inspected visually. The circulating water pumps are tested in accordance with the Hydraulic Institute Standards.

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The condenser is field hydrostatically tested in accordance with the Steam Surface Condenser Standards published by the Heat Exchange Institute.

All major components are inspected and cleaned prior to installation in the system. Preoperational tests are performed after system installation.

Sampling stations and test connections are provided to allow inservice testing during operation of the system.

10.4.5.5 Instrumentation

The circulating water pumps are individually equipped with shutoff valves which are interlocked with their respective pump motors to prevent startup unless the valve is closed and to prevent shutdown unless the valve is less than 15 percent open. Isolation valves are provided at the inlets of each condenser shell which enable any water box to be isolated. The isolation valves are equipped with limit switches and are operated by manual switches located in the main control room. Temperature, pressure, level, conductivity, pH, and chlorine instrumentation are provided to monitor system performance.

10.4.6 CONDENSATE FILTER DEMINERALIZER SYSTEM

10.4.6.1 Design Bases

The condensate filter demineralizer system capacity is 32,000 gpm, which is in excess of 100% rated flow. The 100% rated system capacity is 28,719 gpm.

As a design basis for this system, the effluent water quality is as follows:

Normal Operation	
Specific conductivity at 25°C	0.1 micromho/cm.
pH at 25°C	6.5-7.5
Max. total silica (as SiO_2)	<5 ppb
Max. total iron (as Fe)	<5 ppb
Max. total copper (as Cu)	<2 ppb
Nickel (as Ni)	<2 ppb
Chloride (as Cl)	10 ppb*

* or 25% of influent, whichever is less; min. of 1 ppb.

11.5.2.2.1.7 Radwaste Building Ventilation Release Ducts Radiation Monitoring System

This monitor system measures the radioactivity in the radwaste building ventilation air exhaust as it is being discharged to the environment and in doing so complies with Regulatory Guide 1.21, Rev. 1 and Design Criterion 64. Radioactivity originates from radwaste tank vents, from primary water processing equipment and from laboratory sampling hoods as well as various cubicles having liquid process treatment systems within the building. A continuous sample is drawn from each of the two out of three fan exhausters that are operating. The representative sample is withdrawn through a duct probe, passes through a particulate filter to collect particulates, thence through a charcoal filter to collect halogens. Filters are exchanged at least weekly for laboratory radiochemical analyses. The filter air sample streams are combined to pass through a single gas monitor.

The gas monitor is mounted in a heavily shielded chamber. The gas channel consists of a local detector and preamplifier with countrate meter and a recorder in the main control room.

Arrangement details are shown in Figure 11.5-6.

This monitor has no control functions. There are two adjustable trip circuits, one high for high radiation alarm and one low for instrument inoperative that annunciate in the main control room.

11.5.2.2.2 Liquid Process and Effluent Radiation Monitoring System

These systems monitor gamma radiation levels of liquid process and effluent streams.

Each monitor system consists of a scintillation detector inserted into a well in the process piping or into a sump or in an off-line chamber to which a process stream sample is piped. The detector locations are selected to obtain a reasonable geometry and are positioned away from crud trap and associated high background regions. Full lead shielding is provided around detectors to further reduce background levels except for turbine-generator building drain sump monitors.

For each liquid off-line detector location, a continuous sample is extracted from the liquid process pipe, passed through a liquid sample panel which contains a detection assembly for gross radiation monitoring, and returned to the process pipe. The detection assembly consists of a scintillation detector mounted in a shielded sample chamber equipped with a check source. A radiation monitor in the control room analyzes and visually displays the measured gross radiation level. The sample panel chamber and lines can be drained to allow assessment of background buildup. The panel measures and indicates sample line flow. A solenoid operated check source operated from the control room can be used to check operability of the channel.

Power is supplied from 125 VDC non-divisional buses for the control room radiation monitors and recorders, and from a 120 VAC local bus for the sample panels.

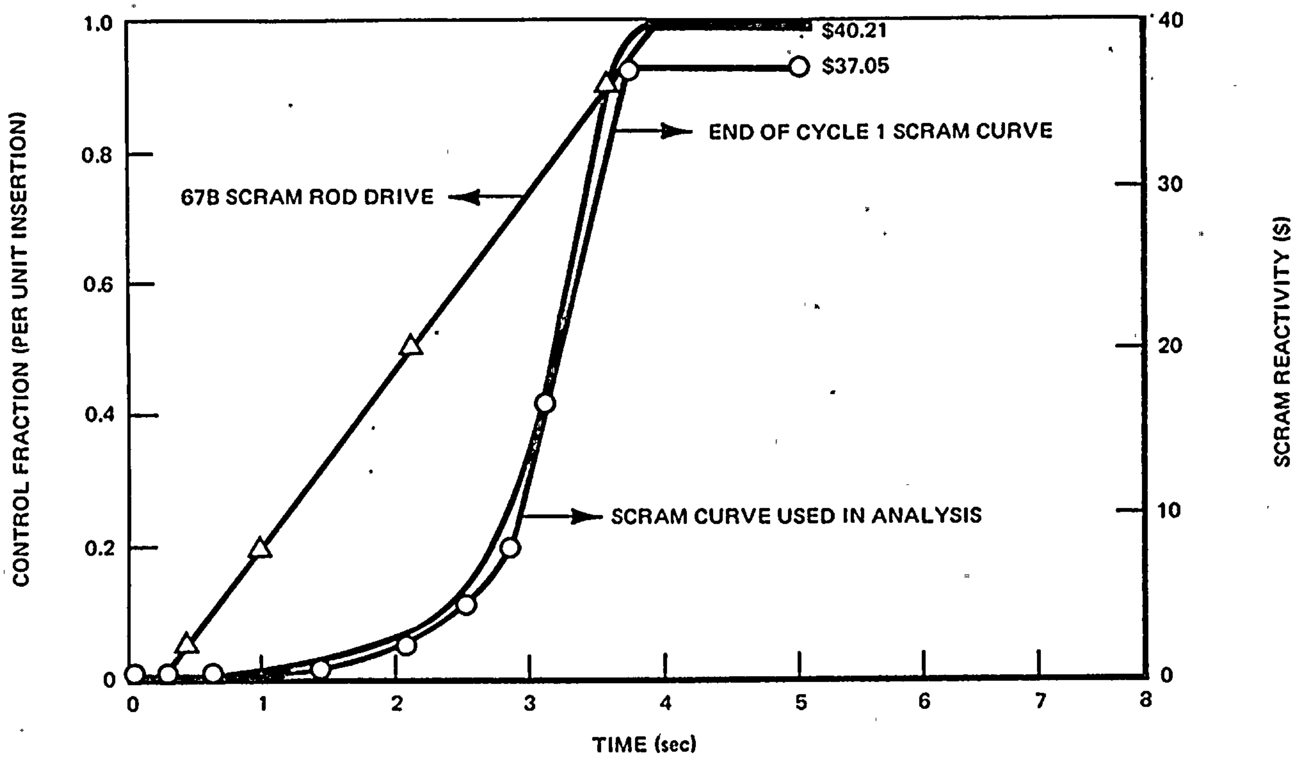
The detector's local preamplifier unit is designed to remain fully operational in their expected environment. If exposed to radiation transients which exceed the channel range, the channel maintains full scale deflection and returns to normal functioning when the transient has subsided.

Each radiation monitor, except for the turbine-generator building sump monitor, has four trip circuits: Two upscale (high-high and high), and one downscale (low), and one inoperative. Each trip is visually displayed on the affected radiation monitor. Two of these trips actuate corresponding control room annunciators: one upscale (high radiation) and the downscale for the affected liquid monitoring channel. High or low sample flow measured at the sample panel actuates a control room high-low flow annunciator for the affected liquid channel.

All alarms are annunciated in the control room. Liquid monitor systems details are given in Table 11.5-2 and the monitor arrangemets are shown in Figure 11.5-7 and 11.5-8.

11.5.2.2.2.1 Standby Service Water Radiation Monitoring System

A radiation detector is located off-line and samples the standby service piping downstream of each of the two residual heat removel (RHR) heater exchanger (loops A and B). These monitors are designed to detect any primary coolant leakage into the standby service water during operation of the RHR heat exchangers in the shutdown heat removal mode.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SCRAM POSITION AND REACTIVITY CHARACTERISTICS	FIGURE 15.0-2

15.1 DECREASE IN REACTOR COOLANT TEMPERATURE

15.1.1 LOSS OF FEEDWATER HEATING

15.1.1.1 Identification of Causes and Frequency Classification

15.1.1.1.1 Identification of Causes

A feedwater heater can be lost in at least two ways:

a. Steam extraction line to heater is closed.

b. Steam is bypassed around heater.

The first case produces a gradual cooling of the feedwater. In the second case, the steam bypasses the heater and no heating of that feedwater occurs. In either case the reactor vessel receives cooler feedwater. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient for analysis considerations. This event has been conservatively estimated to incur a loss of up to 100°F of the feedwater heating capability of the plant and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. The event can occur with the reactor in either the automatic or manual control mode. In automatic control, some compensation of core power is realized by modulation of core flow, so the event is less severe than in manual control.

15.1.1.1.2 Frequency Classification

A feedwater temperature reduction of 100°F has never been reported; therefore, the probability of this event is considered low enough to warrant it being categorized as an infrequent incident. However, because of the lack of a sufficient frequency data base, this transient disturbance is analyzed as an incident of moderate frequency.

This event is analyzed under worst case conditions of a 100°F loss at full power.

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Tables 15.1-1 and 15.1-2 lists the sequence of events for this transient and its effect on various parameters is shown in Figure 15.1-1 and 15.1-2.

15.1.1.2.1.1 Identification of Operator Actions

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. An average power range monitor (APRM) neutron flux or thermal power alarm will alert the operator that he must insert control rods to get back down to the rated flow control line, or that he must reduce flow if in the manual mode. The operator must determine from existing tables the maximum allowable T-G output with feedwater heaters out of service. If reactor scram occurs, as it does in manual flow control mode, the operator must monitor the reactor water level and pressure controls and the T-G auxiliaries during coastdown.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The thermal power monitor (TPM) is the primary protection system trip in mitigating the consequences of this event.

Required operation of engineered safeguard features (ESF) is not expected for either of these transients.

15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The TPM mentioned in 15.1.1.2.2 is the mitigating system and is designed to be single failure proof.

TABLE 15.1-5

SEQUENCE OF EVENTS FOR
INADVERTANT SRV OPENING

Time	Event
0	Initiate opening of 1 safety relief valve which remains open throughout the event.
10 Min	Operator attempts to close valve fail. Operator scrams the plant and MSIV closure occurs (worst case).
10.5 Min	RCIC or HPCS initiate.
20 Min	Operator activates RHR in suppression pool cooling mode.
5 Hours	Shutdown and cooldown completed.

15.1.5 SPECTRUM OF STEAM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT IN A PWR

This event is not applicable to BWR plants.

15.1.6 INADVERTENT RHR SHUTDOWN COOLING OPERATION

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions no conceivable malfunction in the shutdown cooling system could cause temperature reduction.

In startup or cooldown operation, where the reactor is at or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-6.

15.2.3.3.3.2 Turbine Trip with Failure of Bypass

A turbine trip with failure of the bypass system is simulated at 105% NB rated steam flow conditions in Figure 15.2-4.

Peak neutron flux reaches 233.7% of its rated value, and peak fuel center temperature increases approximately 203°F. Since this event is classified as an infrequent incident, it is not limited by the GETAB criteria and the MCPR limit is permitted to fall below the safety limit for incidents of moderate frequency. However, the MCPR for this transient is 1.07 which is just above the safety limit for incidents of moderate frequency and, therefore, the design basis is satisfied.

15.2.3.3.3.3 Turbine Trip with Bypass Valve Failure, Low Power

This transient is less severe than a similar one at high power. Below 30% of rated power, the turbine stop valve closure and turbine control valve closure scrams are automatically bypassed. At these lower power levels, turbine first stage pressure is used to initiate the scram logic bypass. The scram which terminates the transient is initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR remains well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1163 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1136 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1191 psig at the vessel bottom, therefore, the over-pressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed 1163 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since

A reactor scram is initiated by position switches on the stop valves when the valves are less than 90% open. This stop valve scram trip signal is automatically bypassed when the reactor is below 30% NB rated power level.

The analysis presented here is a hypothetical case with a conservative 2 inches Hg per second vacuum decay rate. Thus, the bypass system is available for several seconds since the bypass is signaled to close at a vacuum level of about 10 inches Hg less than the stop valve closure.

15.2.5.3.3 Results

Under this hypothetical 2 inches Hg per second vacuum decay condition, the turbine bypass valve and main steam line isolation valve closure would follow main turbine and feedwater turbine trips about 5 seconds after they initiate the transient. This transient, therefore, is similar to a normal turbine trip with bypass. The effect of main steam line isolation valve closure tends to be minimal since the closure of main turbine stop valves and subsequently the bypass valves have already shut off the main steam line flow. Figure 15.2-6 shows the transient expected for this event. It is assumed that the plant is initially operating at 105% of Nuclear Boiler rated steam flow conditions. Peak neutron flux reaches 157.4% of NB rated power while average fuel surface heat flux reaches 102.6% of rated value. Safety/relief valves open to limit the pressure rise, then sequentially reclose as the stored energy is dissipated.

15.2.5.3.4 Consideration of Uncertainties

The reduction or loss of vacuum in the main turbine condenser will sequentially trip the main and feedwater turbines and close the main steam line isolation valves and turbine bypass valves. While these are the major events occurring, other resultant actions will include scram (from stop valve closure) and bypass opening with the main turbine trip. Because the protective actions are actuated at various levels of condenser vacuum, the severity of the resulting transient is directly dependent upon the rate at which the vacuum is lost. Normal loss of vacuum due to loss of cooling water pumps or steam jet air ejector problem produces a very slow rate of loss of vacuum (minutes, not seconds). See Table 15.2-6. If corrective actions by the reactor operators are not successful, then simultaneous trips of the main and feedwater turbines, and ultimately complete isolation by closing the bypass valves (opened with the main turbine trip) and the MSIVs, will occur.

A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves since they would be closed more quickly.

Other uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves are assumed to be at the upper limit of technical specifications for all valves.

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1162 psig at the vessel bottom. Clearly, the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1135 psig. A comparison of these values to those for Turbine Trip with Bypass, 15.2.3, at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steam line isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under optimum meteorological and release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specifications; therefore, this event, at worst, would only result in a small increase in the yearly integrated exposure level.

path through the RHR shutdown cooling lines. An additional failure is assumed which completely disables the RHR equipment in one division. The operator then establishes a shutdown cooling path for the vessel through the SRV valves.

15.2.9.1.2 Frequency Classification

This event is evaluated as a moderate frequency event. However, for the following reasons it could be considered an infrequent incident:

a. No RHR valves have failed in the shutdown cooling mode in BWR total operating experience.

b. The set of conditions evaluated is for multiple failure as described above and is only postulated (not expected) to occur.

15.2.9.2 Sequence of Events and System Operation

15.2.9.2.1 Sequence of Events

The sequence of events for this event is shown in Table 15.2-12.

15.2.9.2.1.1 Identification of Operator Actions

For the early part of the transient, the operator actions are those described in 15.2.6, loss of offsite power. The operator should re-establish reactor cooling by one or more of the following:

a. Maintain reactor water inventory with the RCIC or HPCS systems.

b. At approximately 10 minutes into the transient, initiate suppression pool cooling (again for purposes of this analysis, "worst case", it is assumed that only one RHR heat exchanger is available).

c. Initiate RPV shutdown depressurization by manual actuation of the safety/relief valves.

d. Attempts to open one of the two RHR shutdown cooling suction valves are assumed unsuccessful (reactor pressure approximately 100 psig).

e. At 100 psig RPV pressure, actuates ADS to complete blowdown; and the operator establishes a reactor cooling path as described in the notes for Figure 15.2-11.

Time required to initiate the necessary steps to maintain reactor pressure and level control is approximately 10 minutes.

15.2.9.2.2 System Operation

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation credit is taken for the plant and reactor protection systems and/or the ESF utilized.

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure (Loss of Division Power) has already been analyzed in this event. Therefore, no single failure or operator error can make the consequences of this event any worse. See 15A for discussion of this subject.

15.2.9.3 Core and System Performance

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. The earliest time the shutdown system can be actuated is 2-3 hours after shutdown is initiated. During this time MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period approximated for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action. Only a qualitative evaluation is provided below since the transient behavior of the core has been evaluated in 15.2.6.

15.2.9.4 Qualitative Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using the redundant shutdown cooling loop. In cases where the RHR shutdown cooling suction line valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-10). An evaluation has been performed assuming a failure that disables the RHR shutdown cooling suction line valves.

This evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (see Reference 15.2-3 and Figure 15.2-11). The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety systems. The systems are capable of bringing the reactor to a cold shutdown in approximately 36 hours or less after the transient occurs.

The systems have suitable redundancy in components such that even for onsite electrical power operation (offsite power is not available) the systems' safety function can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7 psia 200°F) conditions.

15.2.9.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the RCIC/HPCS systems together with the nuclear boiler pressure relief system and the RHR heat exchanger in the suppression pool cooling mode.

For evaluation purposes, however, it is assumed that plant shutdown is initiated by a transient event (loss of offsite power) which results in relief valve actuation and subsequent suppression pool heatup. For this postulated condition, the reactor is shutdown and the reactor vessel pressure is reduced to approximately 100 psig. Manual operation of the safety/relief valves is utilized to depressurize the reactor vessel. Reactor vessel makeup water is automatically provided via the RCIC/HPCS systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC/HPCS and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing reaching the 100 psig level.

15.2.9.4.2 Approximately 100 psig to Cold Shutdown

The following assumptions are used for the analyses of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

a. The vessel is at 100 psig and saturated conditions.

b. A worst-case single failure is assumed to occur (i.e., loss of a division of emergency power) and

c. There is no offsite power available.

In the event that the RHR's shutdown suction line is not available because of single failure, the first action to be taken will be to maintain the 100 psig level while personnel gain access and effect repairs. For example, if a single electrical failure caused the suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. Nevertheless, if for some reason the normal shutdown cooling suction line cannot be restored to service, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34.

TABLE 15.2-12

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

Approximate Elapsed Time	Event
0	Reactor is operating at 105% NBR steam flow when LOP transient occurs initiating plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
10 min.	Controlled blowdown initiated (100°F/hr.) using selected safety relief valves.
15 min.	Suppression pool cooling initiated to prevent overheating from SRV actuation.*
121 min.	Blowdown to 100 psi completed.
121 min.	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
151 min.	Actuate ADS and complete blowdown to suppression pool.
151 min.	Redirect RHR pump discharge from pool to vessel via LPCI line. Alternate cooling path now established.

*See 15.2.6 for detailed sequence of events for loss of AC power transient.

TABLE 15.2-13

INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING

Initial Power Corresponding To	105% Rated Steam Flow
Suppression Pool Mass (lbm)	8.52 E6
RHR (KHX value) (Btu/sec/°F)	289
Initial vessel condition	
Pressure (psia)	1055
Temperataure (°F)	550.7
Initial primary fluid inventory (lbm)	7.016×10^5
Initial pool temperature (°F)	95
Service water temperature (°F)	87
Vessel heat capacity (Btu/lbm/°F)	0.123
HPCS on-off water level (ft.)	
HPCS ON	40.8
HPCS OFF	47
HPCS flow rate (lbm/sec.)	868
LPCI flow rate (lbm/sec.)	982

17.1 QUALITY ASSURANCE DURING DESIGN AND CONSTRUCTION

There are four principal participants in WNP-2 design and construction quality programs. They are the Owner, Washington Public Power Supply System; the Architect/Engineer (AE), Burns and Roe, Inc.; the Nuclear Steam Supply System (NSSS) Supplier, General Electric Company; and the Construction Manager (CM), Bechtel Power Corporation.

a. The Supply System, as the Owner and licensee, has overall responsibility for assuring that the plant is designed and constructed in accord with approved QA programs. The Supply System WNP-2 Project QA organization provides management overview of the other elements of the site QA programs. Section 17.1.1 describes the Supply System WNP-2 Quality Assurance Program.

b. Burns and Roe, Inc. provides Architect/Engineer and related services for WNP-2. Section 17.1.2 describes the Burns and Roe Quality Assurance Program.

c. The General Electric Company provides NSSS design, fabrication, and erection/construction services for WNP-2. Section 17.1.3 describes the GE Quality Assurance Program.

d. The Bechtel Power Corporation provides construction management services for WNP-2. This service consists primarily of direction and coordination of site contractor activities and includes related QA/QC services. Section 17.1.4 describes the Bechtel Quality Assurance Program.

17.1.1 WASHINGTON PUBLIC POWER SUPPLY SYSTEM QUALITY ASSURANCE PROGRAM

The Washington Public Power Supply System (Supply System or WPPSS) has implemented a Quality Assurance Program (QA Program) for the design, procurement, and construction of WPPSS Nuclear Project No. 2 (WNP-2). This QA Program has been implemented in accordance with requirements of Appendix B to 10CFR50. The applicable requirements of Appendix B, 10CFR50 are applied to those items classified as WPPSS Quality Class I due to their relationship to a nuclear safety function.

As the license applicant, the Supply System is responsible for the plant. Therefore, the Supply System WNP-2 QA Program and its implementation has been structured to assure that design, procurement, and construction activities are accomplished in accordance with sound engineering principles and practices. Systems, components, and structures that are safety-related, in the context of 10CFR20, 10CFR50, and 10CFR100, are required to be designed, specified, fabricated, installed, and tested in accordance with applicable regulatory requirements, codes, standards, specifications, and procedures.

The description of the Supply System WNP-2 Design and Constructon QA Program which follows is of the program as it currently exists. This program evolved from the original quality program which first appeared in Appendix D.O of the PSAR. The changes involved in this evolution process include: NRC requested changes; updates in organization responsibilities and authorities; and the incorporation of new requirements.

17.1.1.1 Organization

The Supply System Managing Director is responsible to the Board of Directors for the overall management of Supply System activities, including the establishment and implementation of policies. The Managing Director resolves issues involving quality brought to his attention because of failure to reach resolution at lower levels of management. Overall Supply System organization is shown on Figure 17.1-1.

The Deputy Managing Director is responsible and accountable to the Managing Director for: a) coordinating and integrating the activities of Supply System organizations; b) supporting and advising the Managing Director in his functions of leadership and evaluation; and c) acting for the Managing Director as and when required.

The Quality Assurance Director is responsible and accountable to the Managing Director to develop, administer, and assess the implementation of the Supply System Quality Assurance Program. Included in this responsibility are auditing functions performed on the Supply System WNP-2 quality affecting activities; audits, surveillance or surveys of suppliers of material, equipment, or services for the WNP-2 Project. The Quality Assurance Director has stop work authority. He provides for the review of the status and adequacy of the WNP-2 QA Program on an annual basis.

The Director of Nuclear Safety is responsible and accountable to the Managing Director to develop and administer Licensing,

Operational Nuclear Safety, and Design and Nuclear Safety Assessment activities in support of the Project.

The Technical Director is responsible and accountable to the Managing Director for technical support of Supply System activities from a centralized organization. Functions which are encompassed by this organization include corporate engineering, nuclear fuel management, environmental programs, corporate performance, and project support.

The Director, Contracts and Materials Management is responsible and accountable to the Executive Director for the procurement and control of materials, equipment, and services of the Supply System. This includes the Headquarters and Project Business organizations which provide support to the Program Directors for procurement, contract management, contract administration, including commercial claims analysis and negotiation, business management systems and contract reporting/measurement systems and the Fuel Contracts organization which provides support to the Fuel Supply Department for nuclear fuel procurment and administration. Support is also provided to the Program Director for control of materials, equipment, and services through the Materials Management organization.

The WNP-2 Program Director, as shown on the WNP-2 Project Organizational Chart, Figure 17.1-2, has overall responsibility and authority for all WNP-2 Project activities. He resolves WNP-2 issues involving quality brought to his atten-

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General Compliance or Alternate Approach Assessment (Cont'd.)

1. If Equation 10 of NB-3653-1, ASME Code III results in S < 2.4 Sm for ferritic or austenitic steels, no other requirements need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and an OBE event transient.

2. If Equation 10 results in 2.4 < S < 3.0 Sm for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the basis of Equation 14 of NB-3653.6, must be less than 0.1.

3. If Equation 10 results in S > 3.0 Sm for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must not be greater than 2.4 Sm.

Specific Evaluation Reference:

Refer to 3.6.

Similar Application Reference:

Similar application was utilized on Zimmer and GESSAR.

Regulatory Guide 1.47, Rev. 0, May 1973

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the requirements of IEEE Standard 279-1971 and Appendix B to 10CFR50.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analyis, design and/or equipment utilized on this facility is in compliance with the intent of the regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The system of bypass indication is designed to satisfy the requirement of IEEE 279-1971, Paragraph 4.13 and Regulatory Guide 1.47 and is discussed for each safety-related system, as applicable, under 7.1. The design of the bypass indication system allows testing during normal operation and is used to supplement administrative procedures by providing indications of safety systems status.

The bypass indication system is designed and installed in a manner which precludes the possibility of adverse effects on the plant safety system. The bypass indication system is electrically isolated from the protection circuits such that the failure or bypass of a protective function is not a credible consequence of failures in the bypass indication system and the bypass indication system cannot reduce the independence between redundant safety systems.

Capability exists on a limited basis for status indication at the safety-related equipment subsystem level

General Compliance or Alternate Approach Assessment: (Cont'd.)

within the NSSS scope of supply.

See C.3.0 for further discussion of compliance.

Specific Evaluation Reference:

Refer to 7.1.2.4.

Similar Application Reference:

Similar application was utilized on Zimmer and LaSalle.

Regulatory Gide 1.48, Rev. 0, May 1973

Design Limits and Loading Combinations for Seismic Category I Fluid System Components.

Regulatory Guide Intent:

Regulatory Guide 1.48 delineates acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of the Seismic Category I fluid system components.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design and/or equipment utilized in this facility is in compliance with the intent of the subject regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

For a comparison of NSSS with Regulatory Guide 1.48, refer to the attached tabulation.

The design basis was representative of good industry practices at the time of design, procurement and manufacture and is shown to be in general agreement with requirements of Regulatory Guide 1.48, with the following clarifications:

(a) The probability of an OBE of the magnitude postulated for WNP-2 is consistent with its classification as an emergency event. However, for design conservatism, loads due to the OBE vibration motion have been included under upset conditions, loads due to the OBE vibratory motion plus associated transients, such as a turbine trip, have been considered in the equipment design under emergency conditions consistent with the probability of the

Regulatory Guide 1.47, Rev. 0, May 1973

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

WNP-2 has a divisional system of automatic status indication at either the system or division level which is limited to certain inoperable or bypassed conditions. The status indication system includes the capability for the main control room operator to manually initiate indication of system or division unavailability.

Specific Evaluation Reference:

Refer to 7.1.2.4(e).

Regulatory Guide 1.48, Rev. 0, May 1973

Design Limits and Loading Combinations for Seismic Category I Fluid System Components

Compliance or Alternate Approach Statement:

(To be provided at a later date)

General Compliance or Alternate Approach Assessment:

An investigation of the design limits and loading combinations for all seismic category I fluid system components is presently in progress to determine the extent of compliance with the requirements of Regulatory Guide 1.48. Implementation of this regulatory guide will be discussed when this review has been completed.

Specific Evaluation Reference:

Refer to 3.9.3.1.1.7.

Regulatory Guide 1.64, Rev. 2, June 1976

Quality Assurance Requirements for the Design of Nuclear Power Plants

I. Design and Construction Phase

Compliance or Alternate Approach Assessment:

Regulatory Guide 1.64 Rev. 0 does not apply to WNP-2 since it applies to construction permits docketed prior to September 1973. Regulatory Guide 1.64 Rev. 1 and Rev. 2 do not apply to WNP-2 since they apply to construction permits docketed after April 1, 1975 and July 15, 1976 respectively.

General Compliance or Alternate Approach Assessment:

The Architect/Engineer adopted Regulatory Guide 1.64 Rev. 0. The Design Verification method implemented was primarily the independent review alternative except in limited instances of technical extrapolations such as for the Containment Load Defintion Program.

Appendix B requirements for design control were involved in procurements and construction specfications.

Specific Evaluation Reference:

Not applicable

II. Operatinal Phase

Compliance is discussed in the Topical Report referenced in 17.2

Regulatory Guide 1.67, Rev. 0, October 1973

Installation of Overpressure Protection Devices

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to WNP-2 since the reactor coolant system pressure boundary safety/relief valve relieves to a closed discharge system.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

BWR/3 and /4

Cavitation is prevented by a feedwater flow interlock set at 20% to 30% of normal flow, which reduces pump speed to below 30% of rated. The setpoint is determined by the thermal power level where cavitation would occur when the recirculation pump was operated at rated flow. This is conservative for all lower flows, since NPSH requirements decrease at lower flows, while available NPSH remains essentially the same for any given thermal power level.

BWR/5 and /6

The cavitation characteristics (thermal power level at which cavitation will occur for a given flow) of the jet pumps and recirculation pump are similar to a BWR/3 and /4 except that jet pump cavitation occurs at higher thermal power levels. Protection is provided by measuring the available subcooling and tripping the pumps to 25% speed when there is inadequate subcooling. Temperature elements in the loop suction line and steamline measure the amount of subcooling available to the system.

When the recirculation pump is operating at 100% speed, the flow control valve will cavitate at low flows due to the large pressure drop across the valve. Cavitation protection is provided by a feedwater low-flow control valve position interlock that trips the pumps to 25% speed when there is not enough feedwater flow subcooling to allow operation at low flow control valve positions.

H.1.2.6 Piping

The recirculation system discharge piping configuration was changed to allow room for the flow control valve between the recirculation pump and discharge isolation valve. Pipe diameter was also reduced to minimize drywell space requirements.

H.1.2.7 Recirculation Pump Flow Measurement

The recirculation loop flow element is used to provide indirect core flow indication for the flow bias scram system, to indicate pump performance and jet pump drive flow, and as a feedback variable to the flow control system. In general, these functions do not require high measurement accuracy, although repeatability is required. The flow bias scram system and flow control system require a signal that is proportional to pump flow. For the former system, the signal is used as an indicator of core flow, for the latter, as a feedback signal for maneuvering the flow control valve. The proportionality constant (calibration coefficient - pressure drop versus flow) is unimportant as long as that constant does not change, that is, the element is repeatable.

These functional requirements are satisfied by an elbow flow element where the pressure from the inside to the outside of the elbow is proportional to flow. Consequently, the flow nozzle in the BWR/4 design was replaced by an elbow pressure tap flow element in the recirculation pump suction line for the BWR/5 and /6 design.

H.1.2.8 Recirculation Pump Trip (RPT)

The recirculation pumps are tripped for many reasons, among which are low NPSH, some transients and electrical faults such as short circuits. Only one trip function is currently required to be safety grade, and that function is given the name RPT. The purpose of RPT is to mitigate the thermal consequences of the turbine trip and generator trip transients by tripping the recirculation pumps early in the event, producing rapid pump flow coastdown and additional core voiding, which results in a core reactivity reduction. This system is linked to the reactor protection system (RPS) such that both a scram and a pump trip occur when the turbine stop valves start to close and when turbine governor valve fast closure occurs. Both scram and RPT are bypassed at low thermal power levels.

Since only one power source is available to a BWR/4 pump motor, RPT trips the pumps completely off. The BWR/5 activates the 25% speed source (the low-frequency M-G set) when the pump has coasted down to that speed.

H.1.2.9 Core Flow Measurement

The core flow measurement system is unchanged from the BWR/4 design. For BWR/5 and /6, as an operating convenience, individual jet pump pressure drop signals are fed to the process computer to calibrate the system and obtain the jet pump integrity surveillance data required by the technical specifications.

H.1.2.10 Recirculation System Operation

Due to the changes described in Subsections H.1.2.2 and H.1.2.4, the startup and operation of the BWR/5 recirculation system is significantly changed from previous systems. As a result, new control interlocks were necessary to prevent significant transients, equipment damage, or unnecessary scrams. Electrical interlocks were installed between the LFMG set and the normal power supply to prevent damage to the LFMG set and on the flow control valve to prevent cavitation damage. These interlocks also protect against flow-increase transients when starting the system or transferring to the normal power supply.

LIST OF NRC QUESTIONS

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
005.001	2	7
010.001	1	1
010.002	1	1
010.003	1	1
010.004	1	1
010.005	1	1
010.006	1	13
010.007	1	1
010.008	1	1
010.009	1	1
010.010	1	5
010.011	9	9
010.012	2	5
010.013	2	5
010.014	3	5
010.015	1	13
010.016	3	5,13
010.017	1	13
010.018	1	13
010.019	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
010.020	1	5
010.021	1	13
010.022	1	5
010.023	2	5
010.024	1	5
010.025	1	5
010.026	1	5
010.027	1	5
010.028	2	5
010.029	2	5
010.030	1	5
010.031	1	5
010.032	1	5
010.033	1	5
010.034	1	5
022.001	1	1
022.002	1	1
022.003	1	1
022.004	1	1
022.005	3	8
022.006	1	13

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.007	1	11
022.008	1	14
022.009	1	1
022.010	1	13
022.011	1	1
022.012	1	3
022.013	1	3
022.014	1	13
022.015	1	3
022.016	1	3
022.017	1	3
022.018	1	3
022.019	1	3
022.020	1	3
022.021	1	3
022.022	1	3
022.023	1	3
022.024	1	11
022.025	1	14
022.026	1	3
022.027	1	14

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.028	1	3
022.029	1	3
022.030	1	3
022.031	5	16
022.032	2	5
022.033	1	5
022.034	1	5
022.035	2	5
022.036	1	5
022.037	2	5
022.038	1	5
022.039	2	7
022.040	1	5
022.041	1	5
022.042	1	5
022.043	2	5,14
022.044	1	5
022.045	1	5
022.046	1	5
022.047	1	5
022.048	11	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.049	4	5
022.050	2	5
022.051	1	5
022.052	1	5
022.053	2	20
022.054	2	20
022.055	2	20
022.056	2	20
022.057	1	20
022.058	1	20
022.059	2	20
022.060	1	20
022.061	5	20
022.062	1	20
022.063	3	20
022.107	8	20
022.109	2	20
022.110	1	20
022.111	1	20
022.112	2	20
022.113	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
022.114	1	20
031.001(a)	1	0
031.001(b)	1	14
031.001(c)	1	14
031.001(d)	1	14
031.001(e)	1	0
031.001(f)	1	0
031.001(g)	1	14
031.001(h)	1	14
031.001(i)	2	0
031.001(j)	1	13
031.001(k)	1	13
031.001(l)	1	0
031.001(m)	1	0
031.001(n)	1	0
031.001(o)	1	0
031.001(p)	1	0
031.001(q)	1	0
031.001(r)	2	14
031.001(s)	1	0
031.001(t)	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.001(u)	1	14
031.001(v)	1	14
031.001(w)	1	14
031.001(x)	1	0
031.001(y)	1	14
031.001(z)	1	14
031.001(aa)	1	14
031.001(bb,cc)	1	14
031.001(dd)	1	14
031.001(ee)	1	14
031.001(ff)	1	14
031.001(gg)	1	0
031.001(hh)	1	14
031.001(ii)	1	14
031.002	1	0
031.003	1	0
031.004	1	0
031.005	1	14
031.006	2	14
031.007	1	0
031.008	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.009	4	0
031.010	4	14
031.011	1	14
031.012	1	0
031.013	1	14
031.014	1	3
031.015	3	14
031.016	2	14
031.017	2	0
031.018	2	0
031.019	1	14
031.020	1	0
031.021	2	14
031.022	1	14
031.023	1	0
031.024	1	0
031.025	2	0
031.026	1	10
031.027	1	14
031.028	1	14
031.029	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.030	2	14
031.031	1	14
031.032	1	0
031.033	1	0
031.034	2	0
031.035	1	14
031.036	1	14
031.037	1	14
031.038	2	14
031.039	2	14
031.040	1	14
031.041	1	14
031.042	1	0
031.043	1	0
031.044	2	14
031.045	1	14
031.046	1	14
031.047	1	14
031.048	3	0,14
031.049	1	0
031.050	3	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.051	1	0
031.052	1	14
031.053	2	0
031.054	1	14
031.055	1	13
031.056	1	10
031.057	1	10
031.058	3	11
031.059	2	10
031.060	1	3
031.061	1	3
031.062	1	3
031.063	1	3
031.064	1	3
031.065	1	3
031.066	2	3
031.067	1	3
031.068	1	3
031.069	1	14
031.070	5	3
031.071	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.072	1	14
031.073	1	3
031.074	1	3
031.075	1	14
031.076	3	14
031.077	1	3
031.078	2	3
031.079	2	3
031.080	7	10,13
031.081	2	17
031.082	1	10
031.083	1	10
031.084	1	10
031.085	1	10
031.086	1	10
031.087	5	10
031.088	1	10
031.089	1	10
031.090	1	10
031.091	1	10
031.092	2	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.093	1	10
031.094	1	10
031.095	1	10
031.096	1	10
031.097	1	10
031.098	1	10
031.099	1	10
031.100	25	10,14
031.101	1	10
031.102	1	10
031.103	3	10
031.104	2	10
031.105	2	10
031.106	1	10
031.107	1	10
031.108	2	10
031.109	1	10
031.110	1	10
031.111	1	10
031.112	1	10
031.113	1	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.001	2	0
040.002	1	0
040.003	1	0
040.004	1	1
040.005	1	0
040.006	1	0
040.007	1	0
040.008	1	0
040.009	1	14
040.010	2	0
040.011	1	0
040.012	1	0
040.013	1	0
040.014	1	0
040.015	1	5
040.016	1	0
040.017	1	0
040.018	1	0
040.019	1	0
040.020	7	0
040.021	1	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.022	1	0
040.023	1	0
040.024	2	0
040.025	1	0
040.026	2	5
040.027	1	0
040.028	1	0
040.029	1	0
040.030	1	0
040.031	1	0
040.032	1	0
040.033	1	0
040.034	2	7
040.035	1	7
040.036	2	7,20
040.037	1	7
040.038	1	7
040.039	4	7,11
040.040	1	7
040.041	1	7
040.042	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.043	1	14
040.044	2	7
040.045	5	7,11
040.046	1	7
040.047	1	7
040.048	1	7
040.049	1	7
040.050	1	7
040.051	1	7
040.052	1	7
040.053	2	7
040.054	1	7
040.055	1	7
040.056	1	7
040.057	1	7
040.058	1	7
040.059	1	7
040.060	2	7,14
040.061	1	7
040.062	1	7
040.063	1	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
040.064	1	7
040.065	1	7
040.066	1	7
040.067	1	7
040.068	1	7
040.069	1	7
040.070	1	14
040.071	1	7
040.072	1	7
040.073	1	7
040.074	1	7
040.075	62	20
040.076	4	20
040.077	3	20
040.078	4	20
040.079	10	20
110.001	1	9
110.002	1	9
110.003	1	9
110.004	1	9
110.005	1	9

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
110.006	1	9
110.007	1	9
110.008	1	9
110.009	1	9
110.010	1	9
110.011	1	9
110.012	1	9
110.013	1	9
110.014	1	9
110.015	1	9
110.016	1	9
110.017	1	9
110.018	1	9
110.019	4	9
110.020	1	9
110.021	1	9
110.022	1	9
110.023	1	9
110.024	1	9
110.025	1	9
110.026	1	9

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
110.027	1	9
110.028	2	9
110.029	1	9
110.030	2	9
110.031	1	9
110.032	1	9
110.033	1	9
110.034	1	9
110.035	1	9
110.036	2	9
110.037	2	9
121.001	3	5
121.002	8	5
121.003	1	5
121.004	1	5
121.005	1	5
121.006	1	5
121.007	1	13
121.008	10	5,10
121.009	1	5
121.010	2	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.001	1	1
130.002	1	1
130.003	1	1
130.004	1	1
130.005	1	1
130.006	1	1
130.007	1	1
130.008	1	1
130.009	1	13
130.010	1	8
130.011	1	8
130.012	1	8
130.013	1	8
130.014	1	8
130.015	1	8
130.016	1	8
130.017	1	8
130.018	1	8
130.019	1	8
130.020	1	8
130.021	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.022	1	8
130.023	1	8
130.024	1	8
130.025	1	8
130.026	1	8
130.027	1	8
130.028	1	8
130.029	1	8
130.030	1	8
130.031	1	8
130.032	1	8
130.033	1	8
130.034	1	8
130.035	4	8
130.036	1	8
130.037	1	8
130.038	2	8
130.039	7	8
130.040	1	8
130.041	1	8
130.042	8	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
130.043	1	8
130.044	1	8
130.045	3	12
130.046	3	12
130.047	1	12
130.048	2	12
130.049	3	12
210.001	1	0
211.002	1	8
211.003	1	8
211.004	1	14
211.005	1	8
211.006	5	17
211.007	1	14
211.008	2	14
211.009	1	8
211.010	2	8
211.011	1	8
211.012	2	8
211.013	1	8
211.014	2	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.015	1	8
211.016	3	8
211.017	2	8
211.018	1	8
211.019	3	8
211.020	1	8
211.021	2	8
211.022	1	8
211.023	1	8
211.024	1	8
211.025	1	8
211.026	1	8
211.027	4	8
211.028	2	8
211.029	1	8
211.030	1	8
211.031	4	8,10
211.032	1	8
211.033	4	8
211.034	1	8
211.035	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.036	1	8
211.037	1	17
211.038	3	8
211.039	2	8
211.040	2	8
211.041	1	8
211.042	1	8
211.043	1	8
211.044	1	8
211.045	1	8
211.046	1	8
211.047	1	8
211.048	2	8
211.049	1	11
211.050	1	11
211.051	13	11
211.052	1	11
211.053	1	11
211.054	1	11
211.055	1	11
211.056	2	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.057	1	11
211.058	3	11
211.059	1	11
211.060	1	11
211.061	2	11
211.062	1	11
211.063	1	11
211.064	1	11
211.065	2	11
211.066	2	11
211.067	1	11
211.068	1	11
211.069	1	11
211.070	1	11
211.071	1	11
211.072	1	11
211.073	1	11
211.074	1	11
211.075	1	11
211.076	2	11
211.077	4	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.078	1	11
211.079	4	11
211.080	2	11
211.081	2	11
211.082	3	11
211.083	1	11
211.084	1	11,20
211.085	4	11
211.086	1	11
211.087	2	11
211.088	2	11
211.089	5	11
211.090	1	11
211.091	1	11
211.092	2	11
211.093	1	11
211.094	1	11
211.095	1	11
211.096	1	11,20
211.097	1	11
211.098	1	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.099	1	11
211.100	1	11
211.101	2	11
211.102	1	11
211.103	1	11
211.104	1	11
211.105	3	11
211.106	1	11
211.113	1	20
211.115	1	20
211.124	1	20
211.127	2	20
211.137	1	20
211.139	2	20
211.140	1	20
211.141	1	20
211.142	1	20
211.145	1	20
211.147	1	20
211.148	4	20
211.149	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.151	1	20
211.153	2	20
211.154	1	20
211.155	2	20
211.156	2	20
211.157	2	20
211.158	1	20
211.160	1	20
211.161	1	20
211.162	1	20
211.163	1	20
211.165	2	20
211.166	2	20
211.167	1	20
211.169	2	20
211.170	1	20
211.172	1	20
211.175	1	20
211.176	1	20
211.177	1	20
211.178	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.179	1	20
211.181	1	20
211.183	1	20
211.184	1	20
211.186	1	20
211.187	1	20
211.188	1	20
211.190	1	20
211.191	1	20
211.192	1	20
211.193	1	20
211.194	1	20
211.195	2	20
211.200	1	20
211.205	2	20
211.207	2	20
211.210	1	20
212.001	1	3
212.002	1	3
212.003	2	5,14
212.004	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES.	AMENDMENT
221.001	2	7
221.002	1	7
221.003	1	7
221.004	1	7
221.005	1	7
221.006	1	7
221.007	1	7
221.008	1	7
221.009	1	7
221.010	2	7
221.011	1	7
221.012	3	7
221.013	1	7
222.001	1	8
222.002	3	8
222.003	1	8
222.004	1	8
231.001	1	3
231.002	3	3
231.003	1	5
231.004	2	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
231.005	7	20
232.001	1	3
232.002	1	5
232.003	1	5
232.004	1	5
232.005	1	5
312.001	1	1
312.002	1	1
312.003	1	1
312.004	1	1
312.005	1	1
312.006	1	1
312.007	1	1
312.009	1	8
312.010	1	1
312.011	1	1
312.012	2	1
312.013	1	1
312.014	1	1
312.015	1	8
312.016	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
312.017	1	14
312.018	1	11
312.019	1	5
321.001	1	14
321.002	1	1
321.003	1	13
321.004	1	5
321.005	4	5,20
331.001	1	13
331.002	2	13
331.003	2	1
331.004	1	1
331.005	1	1
331.006	1	1
331.007	1	1
331.008	1	1
331.009	1	1
331.010	1	1
331.011	1	1
331.012	1	1
331.013	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
331.014	1	1
331.015	2	5
331.016	1	5
331.017	1	5
331.018	1	5
331.019	1	5
331.020	1	5
331.021	1	13
331.022	1	5
331.023	1	5
331.024	1	5
331.025	2	20
331.026	2	20
331.027	1	20
331.028	2	20
331.029	1	20
360.001	1	1
360.002	1	1
360.003	1	1
360.004	3	10
360.005	6	10

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362.001	1	3
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362.003	1	3
362.004	2	3
362.005	1	5
362.006	1	5
362.007	1	5
362.008	1	5
362.009	2	5
371.001	1	1
371.002	1	1
371.003	1	1
371.004	1	1
371.005	1	1
371.006	1	5
371.008	1	5
371.009	1	5
371.010	1	5
371.011	1	5
371.012	1	5
371.013	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
371.014	1	5
371.03E	1	20
371.04E	1	20
371.05E	1	20
371.06E	1	20
371.07E	1	20
372.001	1	1
372.002	1	1
372.003	1	1
372.004	1	1
372.005	1	1
372.006	1	1
372.007	1	3
372.008	14	5
372.009	1	5
372.010	1	5
372.011	1	5
372.012	1	5
372.013	1	5
372.014	7	5
372.015	4	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
372.016	5	5
372.017	1	5
421.001	1	20
421.002	1	20
421.003	1	20
421.004	2	20
421.005	1	20
421.006	1	20
421.007	1	20
421.008	1	20
421.009	1	20
421.010	1	20
421.011	1	20
421.012	1	20
421.013	1	20
421.014	1	20
421.015	1	20
421.016	1	20
421.017	1	20
421.018	1	20
421.019	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
421.020	2	20
421.021	1	20
421.022	1	20
421.023	1	20
421.024	1	20
421.025	1	20
421.026	1	20
421.027	1	20
421.028	1	20
421.029	1	20
421.030	1	20
421.031	1	20
421.032	1	20
421.033	1	20
421.034	1	20
421.035	1	20
421.036	1	20
421.037	1	20
421.038	1	20
421.039	1	20
421.040	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
421.041	1	20
421.042	1	20
421.043	6	20
422.001	2	7
422.002	1	7
422.003	1	7
422.004	4	7
422.005	1	7
422.006	3	7
422.007	2	7
422.008	2	7
422.009	1	7
423.001	1	1
423.002	1	1
423.003	1	1
423.004	1	0
423.005	1	1
423.006	2	1
423.007	1	1
423.008	1	1
423.009	1	1

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423.010	1	1
423.011	4	7
423.012	1	7
423.013	1	7
423.014	1	7
423.015	1	7
423.016	2	7
423.017	1	7
423.018	1	7
423.019	7	7
423.020	4	7
423.021	3	7
423.022	1	7
423.023	11	7
423.024	1	7
423.025	1	7
423.026	1	7
423.027	1	7
423.028	1	7
423.029	1	7
423.030	16	20

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423.031	1	20
423.032	1	20
423.033	1	20
423.034	1	20
423.035	1	20
423.036	1	20
423.037	1	20
423.038	1	20
423.039	1	20
423.040	1	20
423.041	6	20
423.042	3	20
432.001	7	5
432.002	2	5
432.003	2	5
432.004	3	5
432.005	1	5
432.006	1	5
432.007	1	5
432.008	1	5
432.009	1	5

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
432.010	1	5
432.011	2	5
432.012	1	5
432.013	2	5
432.014	1	5
432.015	1	5
432.016	1	5
441.001	3	7
441.002	1	7
441.003	1	7
441.004	1	7
441.005	1	7

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022.028	022.057	022.109
022.029	022.058	022.110
022.030	022.059	022.111
022.032	022.060	022.112
022.053	022.061	022.113
022.054	022.062	022.114
022.055	022.063	110.026
022.056	022.107	110.027

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022.034

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031.058	031.001
031.059	031.001
031.060	031.001
031.061	031.039
031.062	031.001
	031.002
	031.023
031.063	031.006
	031.014
	031.016
	031.026
	031.047
031.064	031.009
031.065	031.010
031.066	031.009
	031.018
031.067	031.025
031.068	031.032
031.077	031.021
	031.037
031.078	031.050
031.083	031.006
	031.056
	031.059
031.090	031.032
031.094	031.033
031.097	031.026
031.099	031.030
312.016	022.048
423.011	423.002
	423.006
	423.007
423.024	423.023

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031.001(x)
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040.034
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040.041
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231.005
232.005
312.005
312.015
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331.015
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423.016

Q 010.9

Expand 10.4.5 to provide a detailed evaluation of the effects of a postulated failure of the circulating water system inside the turbine building, including a discussion of the following considerations:

a. The capability of detecting a failure in the circulating water system barrier (e.g., the rubber expansion joints), including a description of the method of detecting such a failure. Identify the design and operating pressures of the various portions of the circulating water system barrier and their relation to the pressures which could exist during malfunctions and failures in the system (e.g., rapid valve closure).

b. The time required to stop the circulating water flow (time zero being the instant failure) including all inherent delays such as operator reaction time, drop out times of the control circuitry and coastdown time.

c. For each postulated failure in the circulating water transport system barrier, provide the rate of rise of water in the associated spaces and the total height of the water when the circulating water flow either has been stopped or overflows to the site grade.

d. For each potentially flooded space, provide a discussion and drawing of the protective barrier provided for all safety-related systems that could be affected in the event of flooding. Include in your discussion the consideration given to passageways, pipe chases, and/or the cableways connecting the flooded spaces to the spaces containing safety-related systems or components outside the turbine building. Discuss the effect of the flood water on all potentially submerged safety-related electrical systems and components.

Response:

Please see expanded 10.4.5.3 for the requested evaluation.

Q 010.10
(3.4.1)

Demonstrate that all piping and electrical penetrations in safety-related structures that are below the level of the Probable Maximum Flood are water-tight.

Response:

As stated in 3.4.1.4.1 the plant site grade is higher than the design basis flood elevation resulting from the probable maximum precipitation (PMP) event. Due to the short duration of the PMP flood, the ground water level at the plant site is not affected. As stated in 3.4.1.4.2, piping and electrical penetrations are above the design basis groundwater level and are therefore not sealed against groundwater pressure.

Q. 010.021
(9.1.3)

Provide a cooling system and a source of makeup water for the spent fuel pool which are both designed to Seismic Category I criteria in accordance with the staff positions contained in Regulatory Guide 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis," December 1975.

Response:

WNP-2 has a Seismic Category I source of makeup water for the spent fuel pool from the Seismic Category I standby service water service system. This is shown on Figure 9.1-4 and stated in 9.1.3.3. Cooling under emergency conditions for the fuel pool is supplied by evaporation of pool water. Regulatory Guide 1.13, Rev. 1, makes no specific statements about requiring a Seismic Category I spent fuel pool cooling system. As a result, WNP-2 meets the applicable criteria of the Regulatory Guide and the intent of the question. However, further evaluation of the design in this area is ongoing due to the interaction of fuel pool cooling and post-LOCA secondary containment pressure-temperature response. (See the response to Question 312.018).

Q 010.22
(9.2.1)

Identify which valves are used to isolate that portion of the plant service water system which is not designed to Seismic Category I criteria from that portion which is designed to these criteria. Provide a failure modes and effects analysis for the plant service water system, assuming a seismic event has occurred.

Response

The plant service water system (TSW) is not required for safe shutdown and accordingly is not designed to Seismic Catetory I requirements. A failure modes and effects analysis is not considered necessary. The portions of the TSW system piping in the Reactor Building have been designed to Seismic Category I requirements so that they will not fall and damage safety related equipment.

The standby service water system (SW) is used for safe shutdowns and is designed to Seismic Category I criteria. The SW is discussed in 9.2.5. The plant service water system (TSW) and the standby service water system (SW) are independent systems and are not connected, therefore there are no valves which are used to isolate these systems from each other.

Q. 010.27
RSP
(9.2.7)

We require that you protect the standby service water system from tornado missiles.

Response:

The standby service water system (except for the spray pond spray piping) is protected from tornado missiles. The structures which house the standby service water systems (Reactor Building, DG Building, Control Building, and SW Pumphouse) have been designed to withstand design basis tornado generated missiles as described in 3.5.1.4.1.

Buried portions of the standby service water system are protected from tornado missiles as described in 3.5.2.

See the response to question 10.24 as to why it is not necessary to protect the spray pond spray headers from tornado missiles.

Q. 010.28
(9.3.4)

Describe how flooding of safety-related equipment due to backflooding through the equipment and floor drainage system, is prevented. Demonstrate that those portions of the drainage system necessary to prevent backflooding (e.g., check valves) are designed to Seismic Category I criteria and that their system function will be maintained, assuming a single active failure.

Response:

It is assumed that the question is directed to FSAR 9.3.3.2.2.1, Reactor Building Floor Drains, and not 9.3.4, Chemical and Volume Control System.

As shown on Figure 9.3-8, the floor drain piping in the reactor building drains to one of four sumps listed below.

FLOOR DRAIN SUMP	ROOM LOCATIONS	ROOMS SERVED
FDR-R-1	RHR A Pump Room	RCIC RHR A
FDR-R-2	RHR B Pump Room	RHR B
FDR-R-3	HPCS Pump Room	HPCS CRD
FDR-R-4	RHR C Pump Room	LPCS RHR C

Each of the four downcomers is equipped with instrumentation which alarms in the control room to tell the operator at which elevation an excess of water is collecting in the downcomer. Each sump is equipped with level instrumentation which: 1) controls the sump pumps, 2) alarms in the control room (on high sump level), and 3) initiates closure of the isolation valves in the downcomers and in the piping between interconnected rooms. Not currently shown on Figure 9.3-8 are Class IE level instrumentation to be installed just above floor level in each ECCS pump room. This instrumentation will alarm in the control room.

Q 010.33
(9.4.7)

Provide your analysis which demonstrates that the potential failure of the heaters in the Diesel Generator HVAC System which are not designed to Seismic Category I criteria, will not have an adverse affect on the functional capability of either the Diesel Generator or the Diesel Generator HVAC System.

Response

There are two types of heaters in the diesel generator spaces, electric unit heaters in the diesel oil pump rooms and electric heating coils in the duct systems in the diesel engine rooms themselves. The electric unit heaters in the pump rooms are Seismic Category II. These heaters are supported as Seismic Category I, however, and can fail in place without affecting any safety related equipment. Oil pump room unit heaters are used only for maintenance during cold weather for personnel comfort. The heating coils in the generator room themselves are Seismic Category I and are designated Class 1E.

Q. 010.34
(10.4.5)

Your response to Item 010.09 is unacceptable. Specifically, your analysis of flooding due to failure of the circulating water system is based on a crack whose area is equal to one-quarter of the pipe diameter times the pipe thickness (.5t X .5d). Provide an analysis of flooding due to a postulated failure of the expansion joint in the circulating water system assuming a double-ended guillotine break at this location.

Response:

The original response to Item 010.09 has been rewritten for clarity (see 10.4.5).

The double-ended guillotine break referred to above was not considered. The circulating water system is a moderate energy system by definition. Therefore, in accordance with NRC Standard Review Plan Section 3.6.1, 3.6.2, and 10.4.5, and the associated Branch Technical Position MEB 3-1, the criteria for a postulated failure shall be a through - wall leakage crack of the type addressed in the written response (10.4.5). In any case, as stated at the end of 10.4.5, circulating water piping is located remote from any safety-related equipment. The piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, safety-related equipment is not vulnerable to environmental effects of a circulating water pipe rupture. The pipe exits the room below grade in its routing to and from the cooling towers. It should be also noted that the condenser is located on grade level. Therefore, water above the floor elevation will drain outside and not collect other than in collection basins.

AMENDMENT NO. 20
October 1981

The information on this page is proprietary and has been submitted under separate cover.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	QUENCHER LOCATION FOR TWO VALVE DISCHARGE CASE	FIGURE 022. 062-1

References:

1. Bedrosian, B., "Analysis of a Mark II Containment Structure for Hydrodynamic Loads in Suppression Pool", Proceedings, Conference on Structural Analysis, Design, and Construction in Nuclear Power Plants, Vol. 2, Porto Alegra, Brazil, April 1978.

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A point by point discussion summarizing the WPPSS design capabilities to mitigate Bypass Leakage problems based on the above correspondence and with respect to the proposed Branch Technical Position is given below:

1. NRC Proposed Requirement: Allowable bypass capability on the order of 0.05 ft^2 ($A/\sqrt{K}$)

   Response: As documented in Reference 5 and the FSAR, the maximum allowable bypass leakage capacity is $A/\sqrt{K} = 0.028$ ft^2 using conservative calculational techniques and assumptions.* WPPSS, therefore, believes the existing calculations meet the intent of $A/\sqrt{K} = 0.05$ ft^2.

2. NRC Proposed Requirement: An automatic system should be provided to initiate automatic wet-well sprays. The system should meet the standards of an Engineering Safety Feature including redundancy and diversity and be actuated automatically ten minutes following a LOCA. If the RHR system is used for this purpose, it must be analyzed to assure no degradation of its ECCS function.

   Response: WPPSS asserts that manual initiation is sufficient since the drywell floor will be routinely tested and evaluated against a Tech Spec limit of $A/\sqrt{K} = 0.0045$ ft^2, a level at which no operator action is required for the spectrum of small break sizes. (Reference 5 - see #3 below for testing details).

   The construction, design, quality control, and surveillance requirements on the drywell floor give it the same level of safety as the containment itself. Reference 4 and Part VI of Reference 2 showed that through-wall cracks will not develop through the concrete slab under postulated design conditions including

* The FSAR currently lists the capability as 0.026 ft^2. This is from a GE analysis and the FSAR is being amended to reflect the latest calculations.

the SSE and that leakage in excess of that accounted for due to permeability would not be possible. Reference 6 indicated the NRC Structural Engineering Branch's acceptance of these responses. Accordingly, WPPSS sees no reason to assume that an $A/\sqrt{K}$ of 0.0045 ft^2 is exceeded any more than there would be reason to assume the design containment leak rate of .5% per day is exceeded. Calculations documented in Reference 5 using the CONTEMPT - LT computer code were used in computing the maximum allowable leakage rate of $A/\sqrt{K} = 0.028\ ft^2$, six times the Tech Spec limit. In the calculation over 167 minutes was available for operator action before drywell design pressure was exceeded. Accordingly, a requirement that an automatic system be provided is unnecessary.

3. NRC Proposed Requirement: A single peroperational high pressure leakage test should be performed and periodic low pressure tests at each refueling outage with an acceptance criterion of 10% of the bypass capability at the test pressure.

Response: The intent of this proposed requirement has been committed to by WPPSS. A single preoperational leakage test will be conducted with the downcomers capped at 15 psid and 25 psid (the design drywell to wetwell differential pressure). At each refueling outage a low pressure operational test will be performed as a Tech Spec Surveillance Requirement to verify 0.0045 ft^2. Details of the nature of this test are discussed in question 5.22 to the PSAR but will be summarized here since the specific numbers have been since updated.

Routine Leak Testing and Inspection: During entry to the drywell at each refueling outage, accessible drywell to wetwell barrier surfaces will be visually inspected to ascertain any possible leak paths. Vacuum relief valves will be visually inspected to insure they are clear of foreign material. At each refueling

outage, before the primary system is pressurized, after all these containment inspections are complete, and after the vacuum breakers are exercised, the following test shall be carried out:

> The drywell will be pressurized to at least 1.0 psi above the wetwell. After an adequate stabilization period, the drywell to wetwell leakage rate will be measured. The acceptance criterion will correspond to an equivalent leakage capacity ($A/\sqrt{K}$) of 0.0045 ft^2, which is 16% of the allowable leakage. If a greater leakage rate is found, the containment shall be entered and the cause determined and corrected and the test repeated.

4. NRC Proposed Requirement: Vacuum relief valves should have redundant position indicators with indication and redundant alarms in the control room. The vacuum breakers should be operability tested at monthly intervals to assure free movement.

   Response: WPPSS meets this requirement with the current design. Each vacuum breaker penetration consists of two discs in series, each disc with redundant position indication which display in the Control Room. Each vacuum breaker disc will be equiped with an exercising mechanism and each disc will be exercised at a frequency equivalent to the testing of ECCS valves.

References

1. Letter, WR Butler, NRC, to JJ Stein, WPPSS, "Meeting Summary", October 17-18, 1973, dated November 26, 1973.

2. Letter, WPPSS to NRC, GO2-74-17, dated August 9, 1974.

3. Letter, NRC to WPPSS, dated January 14, 1975.

4. Letter, WPPSS to NRC, GO2-75-52, dated February 25, 1975.

5. Letter, WPPSS to NRC, GO2-76-156, dated April 23, 1976.

6. Letter, NRC to WPPSS, dated May 15, 1975.

Q. 031.113
RSP
(15.4.1.2)

It is our position that the rod sequence control system does not satisfy the requirements of IEEE Standard 279-1971 and, therefore, is unacceptable for the prevention of a control rod withdrawal accident. Accordingly, we require you to provide a modified design for the WNP-2 facility.

Response:

See response to Question 031.098.

Q. 040.55
(9.5.5)

In 8.3.1.1.8.1.11 of the FSAR, you state that the WNP-2 diesel-generators are automatically started and then run without any electrical loads following a postulated LOCA if offsite power is available at the 4.16kV Class 1E busses. Discuss how long the Division 1 and 2 diesel-generators can run unloaded without a degradation of the diesel engine performance or without a loss of reliability. Provide a similar discussion for the Division 3 diesel generator which supplies power to the high pressure core spray system.

Response:

Section 8.3.1.1.8.1.11 of the FSAR for the Division 1 and 2 diesels and 8.3.1.8.2.11 of the FSAR for the HPCS diesel refer to lightly loaded conditions of less than 50% of nameplate rating. Lightly loaded implies loading conditions of 0% to 50%, including the unloaded condition.

As stated in the above referenced sections, the Division 1, 2 and 3 diesel-generating units can run for four (4) hours in the unloaded condition without a degradation of the diesel engine performance or without loss of reliability. Administrative procedures are employed to manually load the diesels to meet the necessary limit.

Q. 40.56
(9.5.5)

In 9.5.5.2 of the FSAR, you state that each diesel engine cooling water system is provided with an expansion tank to provide for system expansion and for venting air from the system. In addition, the expansion tank is intended to: (1) provide for minor system leaks at pump shaft seals, valve stems and other components; and (2) maintain the required net positive suction heat (NPSH) at the cooling water system circulating pump. Indicate the size of the expansion tank and state its location. Demonstrate by analysis that the size of the expansion tank is adequate to: (1) maintain the required NPSH at the pump; and (2) provide makeup water for seven days of continuous operation of the diesel engine at its full rated load without the addition of water into the expansion tank. Alternatively, provide a Seismic Category I Safety Class 3 makeup water supply for the expansion tank.

Response:

The 94 gallon expansion tank is mounted on the diesel engine skid approximately 20" above the cooling water system circulating pumps.

Diesel-generator unit reliability including the functions required of the circulating water pump and expansion tank were demonstrated prior to installation (see 8.3.1.1.9). Periodic testing and maintenance assure continued reliability.

The expansion tank is provided with a level sight glass which is mounted on the front with instructions that indicate minimum water level. An alarm is provided in the control room to annunciate in case of low water level. A Seismic Categroy I Safety Class 3 connection will be provided so water can be supplied from the Seismic I standby service water system to the expansion tank.

AMENDMENT NO. 9
April 1980

Q. 110.033
(3.6.2)
(3.9.3)

For ASME Class 1, 2 and 3 components that could be exposed to either jet impingement loads or to pipe whip impact loads resulting from postulated pipe breaks in adjacent high energy piping, describe how you determine the stress levels in the targeted components. In your response, include a discussion of the structural effects throughout the targeted system from the loads cited above (i.e., those loads associated with postulated pipe breaks) in combination with other applicable loads. Provide assurance that the calculated stress levels are kept below the Service Level D limits of Section III of the ASME Code. If applicable, more conservative limits on stress levels should be imposed for active components or where piping functional capability is required.

Response:

The complete response to NRC Question 110.033 awaits the results of the ongoing pipe break and missile study, and will be presented in a future amendment to the FSAR.

At this time it is anticipated that if the availability of ASME Section III Class 1, 2 and 3 components is needed to safely shut down the plant, and if those components are exposed to postulated jet impingement loads, pipe whip impact loads, and missile loads protective measures will be taken to preclude such loading. It if is determined that protective measures are not required on the basis that the calculated stress levels due to the postulated pipe break for missile loads in combination with other applicable loads are kept below the Service Level D Limits of ASME Section III, such structural analysis will be in accordance with industry-accepted methods. If this approach is used, the methodology and results will be reported in the FSAR.

AMENDMENT NO. 9
April 1980

Q. 110.034
RSP
(3.9.6)

In accordance with 10CFR50.50a(g), we require submittal of your program for inservice testing of ASME Class 1, 2 and 3 pumps and valves. Our positions on this matter are presented in Section 3.9.6 of the SRP. Appendix C to Section 110 provides a suggested format for this submittal and includes a discussion of the information we require to justify any requests for relief from our positions on this matter.

Response:

In accordance with 10CFR50.55a(g), we will prepare a program for operational readiness testing of ASME Class 1, 2 and 3 pumps and valves. Our intentions are to submit this program for your review approximately one year prior to scheduled fuel load. Our program will be in agreement with your positions as presented in Section 3.9.6 of the SRP and the program will be submitted in a format similar to the one described in Appendix C to Series 110 questions.

Q. 211.003
(5.2.5)

The design of the WNP-2 facility routes drainage of both "hot" and "cold" reactor coolant leakage into the drywell equipment drain sump. However, relatively hot sources of leakage water (e.g., the reactor vessel head flange, the vent drain and valve packings) may flash into steam which then must be condensed before it can be drained into the sump. Accordingly, indicate what assurance there is that this steam from relatively hot sources will be condensed so that monitoring of this leakage may be performed in the drywell sump to detect these "hot" leaks. Since leakage from those sources which are relatively cool is drained into the floor drain system, this system should be tested periodically for blocked lines. Accordingly, discuss the surveillance program you propose for the WNP-2 facility to detect blockage of the floor drain system.

Response:

WNP-2 routes "hot" reactor coolant leakage to the drywell equipment drain sump through leakoff lines to a main header. The header discharge is routed to the equipment drain condenser where "hot" leakage is cooled and condensed. Discharge from the condenser to the sump is through a quencher. This arrangement assures that any "hot" leakage which could flash into steam will be cooled and condensed before entering the open sump.

The floor drain sump drain line from the containment to the reactor building sump will be periodically back flushed to show that the drain line is not blocked.

Q. 211.004
(5.2.1)
(7.6.2)

In conformance with the staff's position in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973, you state in Section 7.6.2.4 of the FSAR that the radioactivity monitoring channels are qualified for operation following a Safe Shutdown Earthquake (SSE). Confirm that all of the other leakage detection methods and/or systems will function properly following an Operating Basis Earthquake (OBE). These other leakage detection systems include the drywell equipment sump and the floor drain sump, the sump coolers, and the associatd instrumentation and piping.

Response:

The WNP-2 drywell floor drain and equipment drain sumps, piping to the sumps, and the equipment drain cooler are seismically supported such that they will continue to pass leakage flow following an OBE.

The flow monitoring instrumentation is not qualified to withstand an OBE. WNP-2 will initiate a design change to qualify the existing flow monitoring instrumentation to assure its accuracy following an OBE.

Q. 211.005
(5.2.5)

Provide a list of the normal and the maximum anticipated leakage rates through the reactor coolant pressure boundary (RCPB), including the concentrations of radioactivity in this leakage flow, from both identified and unidentified sources (e.g., the control rod drive flanges and the vent cooler drains) which are routed into the drain sumps.

Response:

The normal and maximum anticipated leakage rates from the reactor coolant pressure boundary within the drywell during normal reactor operation is between 0.1 and 0.5 gpm for unidentified leakage and 2 to 5 gpm for identified leakage. These leakage rates are measured at the sumps and are based on operating plant experience.

The activity concentrations in this leakage are expected to be the same as those in the reactor water. The concentrations are given in Chapter 11, Tables 11.1-2, "Halogen Radioisotopes in Reactor Water"; 11.1-3, "Other Fission Product Radioisotopes in Reactor"; 11.1-4, "Coolant Activation Products in Reactor Water and Steam"; and 11.1-5, "Noncoolant Activation Products in Reactor Water".

Q. 211.006
(5.2.5)
(7.6.2)
(12.3.4)

Provide a detailed discussion of the sensitivity and response times, of the containment airborne radiation monitoring systems for a number of containment background activity levels. The background activity levels which should be considered are those levels in the containment that would result from leakage through the RCPB assuming: (1) relatively clean water in the reactor coolant system at the initial operation of the WNP-2 facility at power; and (2) the maximum level of activity in the reactor coolant permitted by the WNP-2 Technical Specifications. In responding to this item, assume both the normal and the maximum leakage rates identified in your response to Question 211.005. Indicate your assumptions in estimating the response times of the containment airborne radiation monitoring systems (e.g., the preset alarm level for higher background leakage and the plateout factor).

Response:

The response to this question is pending on the sensitivity curves which are to be prepared by the equipment vendor, Kaman Sciences Corporation. These curves have not yet been received by WPPSS. The submittal date to the NRC of the final response to this question will depend on the receipt of the aforementioned curves.

Q. 211.005
(5.2.5)

Provide a list of the normal and the maximum anticipated leakage rates through the reactor coolant pressure boundary (RCPB), including the concentrations of radioactivity in this leakage flow, from both identified and unidentified sources (e.g., the control rod drive flanges and the vent cooler drains) which are routed into the drain sumps.

Response:

The normal and maximum anticipated leakage rates from the reactor coolant pressure boundary within the drywell during normal reactor operation is between 0.1 and 0.5 gpm for unidentified leakage and 2 to 5 gpm for identified leakage. These leakage rates are measured at the sumps and are based on operating plant experience.

The activity concentrations in this leakage are expected to be the same as those in the reactor water. The concentrations are given in Chapter 11, Tables 11.1-2, "Halogen Radioisotopes in Reactor Water"; 11.1-3, "Other Fission Product Radioisotopes in Reactor"; 11.1-4, "Coolant Activation Products in Reactor Water and Steam"; and 11.1-5, "Noncoolant Activation Products in Reactor Water".

Q. 211.006
~~(5.2.5)~~
(7.6.2)
(12.3.4)

Provide a detailed discussion of the sensitivity and response times of the containment airborne radiation monitoring systems for a number of containment background activity levels. The background activity levels which should be considered are those levels in the containment that would result from leakage through the RCPB assuming: (1) relatively clean water in the reactor coolant system at the initial operation of the WNP-2 facility at power; and (2) the maximum level of activity in the reactor coolant permitted by the WNP-2 Technical Specifications. In responding to this item, assume both the normal and the maximum leakage rates identified in your response to Question 211.005. Indicate your assumptions in estimating the response times of the containment airborne radiation monitoring systems (e.g., the preset alarm level for higher background leakage and the plateout factor).

Response:

The two types of radiation monitors used in the WPPSS Nuclear Project No. 2, for monitoring the drywell atmosphere, are the particulate monitor and the noble gas monitor. The sensitivity of these two types of monitors is given in Figure 211.006-1 and 211.006-2. The same detector is used in both monitors. The detector's noise level is about 25 cpm. The minimum detectable concentration is based on doubling the background count rate. The count ratemeter range and the minimum sensitivity of both types of monitors are:

Noble Gas

Count Ratemeter Range:

1.4×10^{-7} μci/cc - 1.4×10^{-1} μci/cc for Kr-85

Minimum Detectable Concentration:

3.6×10^{-7} μci/cc for Kr-85

Particulate

Ratemeter Range:

2.9×10^{-12} μci/cc - 2.9×10^{-6} μci/cc for Sr/Y-90

Minimum Detectable Concentration:

7.4×10^{-12} μci/cc for Sr/Y-90

The quantity of drywell atmosphere that flows through the filter of the particulate monitor and then through the 2.2 liters chamber of the noble gas monitor is 3 cfm. The drywell atmosphere sample is returned to the drywell. (The free volume of the drywell is 200,540 ft^3). There is a charcoal filter after the particulate and before the noble gas detector. The filter efficiency of the particulate filter is assumed to be 100 percent. Similarly, the efficiency of the charcoal filter is assumed to be 100 percent for all halogens.

The 2-inch diameter scintillation crystal, of the particulate detector, is 1/4-inch away from the face of the filter tape. The filter tape is 2.5 inches wide and moves at 1"/hr.

As per the response to Question 211.005, the total minimum identifiable and unidentifiable leakage rate is taken to be 2.1 gpm. The total maximum identifiable and unidentifiable leakage rate is taken to be 5.5 gpm. The leakage is measured at drywell environmental conditions and thus the water density is assumed to be 1g/cc. Based upon potential sources of leakage (such as number and nominal size of valves), it is estimated that 38.5 percent of the leakage is due to steam leakage. Collection of the identifiable leakage is not seal-tight and thus, volatile radioisotopes can escape into the drywell atmosphere. Water leakage is assumed to be flashing, and that there is an instantaneous mixing of the volatile radioisotopes with the drywell atmosphere. It is assumed that the noble gases are in the steam phase and the small quantities of noble gases in the liquid phase are neglected. On the other hand, it is assumed that the quantities of particulate radioisotopes in the steam phase are small and thus, are neglected.

Decay was considered for the radioisotopes in the drywell and for those radioisotopes accumulated on the particulate filter. No decay was considered, however, for the radioisotopes while in transit to the detectors and while in the noble gas detector chamber (for the atmosphere in the chamber, exchange rate is 1.55 seconds). It is assumed that plating, settling,

impingement, etc., reduce the specific concentration of particulate isotopes in the drywell atmosphere by a factor of 1000 before the air flow reaches the filter of the particulate monitor.

In order to be responsive to the question, the source concentrations used in this analysis were taken to be those in Table 5 of ANS/ANSI N237-1976, reduced by 1/100, as representative of "relative clean water in the reactor coolant". The design basis concentrations, as per General Electric specification document No. 22A2703F, Revision 3, were used as representative of the maximum expected level of radioactivity within the reactor coolant.

The criterion used, as an indication of leakage increase, is the doubling of the background count rate within one hour for 1 gpm (additional) leakage. Each detector was evaluated with respect to responding to the criterion. See Table 211.006-1 for the results of the analysis.

In summary, the particulate monitor will meet the requirements stated in Regulatory Guide 1.45 for the minimum activity concentration of radioisotopes in the reactor coolant. For the cases where it is assumed that the design basis activity exists in the reactor coolant, the background activity exceeds the particulate monitor range. However, the detector can be desensitized accordingly. It can be shown that if a reactor coolant activity is selected based upon the guidance contained in Regulatory Guide 1.45; i.e., if "a realistic primary coolant radioactivity concentration" is used, e.g., equal to that given in Table 5 in ANS/ANSI N237-1976, and expanding the criterion of double background count rate to the desensitized monitor, the requirements stated in the regulatory guide will be met.

The noble gas monitor, however, even though its sensitivity is consistent with Regulatory Guide 1.45 requirements, would not be capable of detecting the additional 1 gpm leakage within one hour utilizing the criterion of double background count as positive indication. The noble gas monitor does, however, provide the most reliable and fastest means of ascertaining increased activity within containment with unidentified leakages higher than 1 gpm.

TABLE 211.006-1

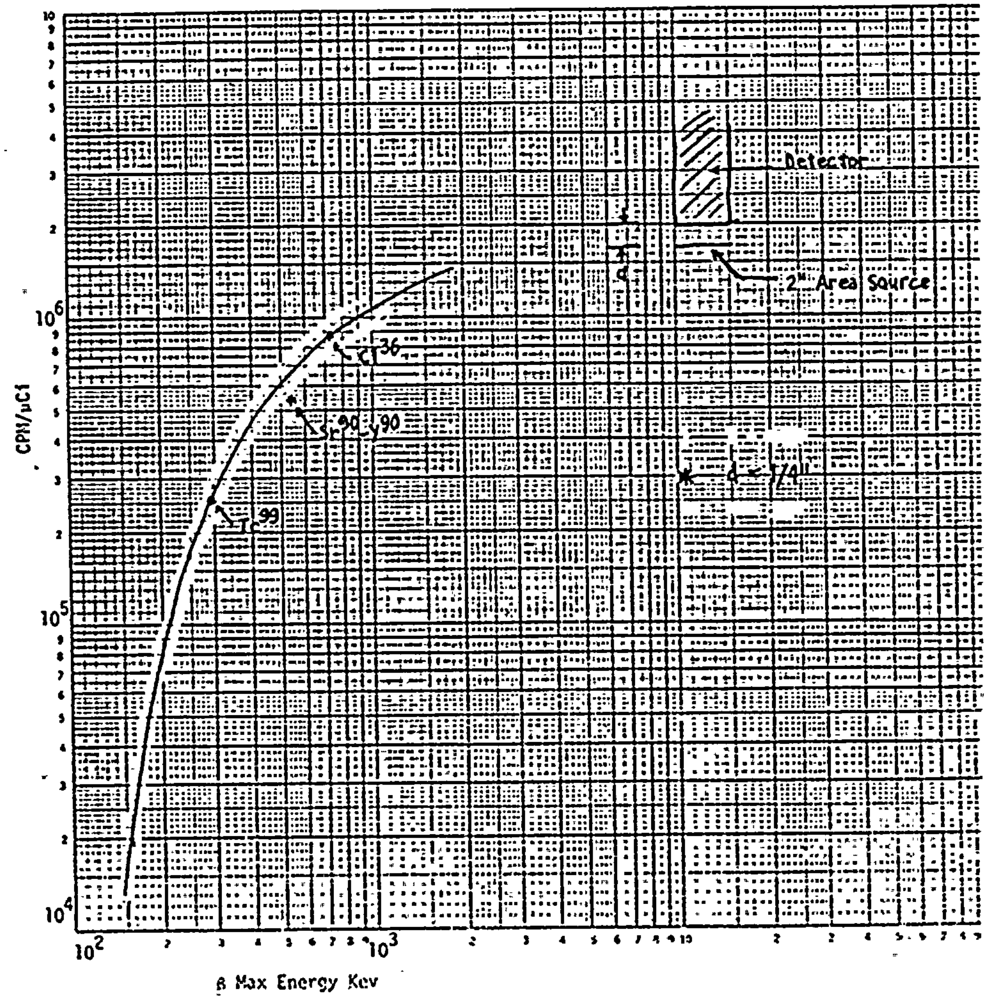
RESULTS OF MONITOR ANALYSIS

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum back-ground leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 5 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector before the event is about 30. With the increase of 1 gpm leakage, the count rate would increase one hour after the event to only about 32 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 46 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 12 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector before the event is about 37. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only 39 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 49 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 81 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 106. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 136 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 51 minutes for the back-ground count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of back-ground leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 213 cpm is added to the detector's noise level of ~25 cpm, the total cpm by the detector, before the event, is about 238. With the increase of 1 gpm leak-age, the count rate would increase, one hour after the event, to only about 269 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 53 minutes for the back-ground count rate to double.

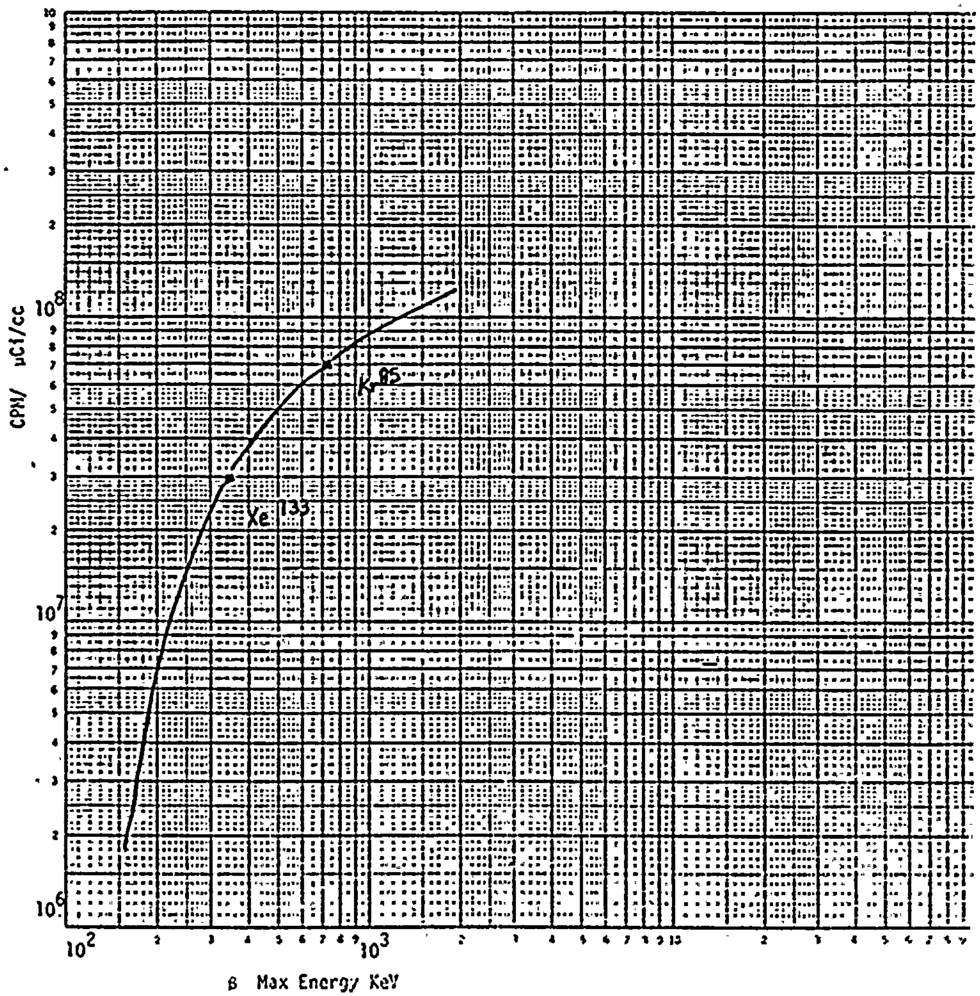
TABLE 211.006-1 (Continued)

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 10 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 35. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 36 cpm.	The background count rate will double as a result of 1 gpm unidentified leakage, within abut 51 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 25 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 50. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 52 cpm.	The background count rate would double as a result of the 1 gpm unidentified leakage, within about 53 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duraton of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 165 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 190. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 219 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 57 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 431 cpm is added to the detector's noise level of ∿25 cpm, the total cpm by the detector, before the event, is about 456. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 485 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 58 minutes for the background count rate to double.

AMENDMENT NO. 17
July 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PARTICULATE SYSTEM EFFICIENCY	FIGURE 211. 006-1



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GASEOUS EFFLUENT SYSTEM EFFICIENCY	FIGURE 211. 006-2

Q. 211.007
(7.6.2)

Expand your discussion in Section 7.6.2.4 of the FSAR regarding the operability verification and the calibration of the leakage detection system which will be accomplished by comparing the results of diverse monitoring methods. For example, if a radioactivity monitoring system is checked against the sump level and the flow monitoring system, indicate how the latter system is determined to have acceptable accuracy. Confirm that calibration and operability tests will: (1) be performed periodically during plant operation; and (2) be in compliance with the requirements of IEEE Standard 279-1971.

Response:

Comparisons between the drywell floor drain flow monitoring system and radioactivity monitoring systems will be made during plant operation. The sensitivity of the monitors are within the guidelines of Regulatory Guide 1.45 (May, 1973) whereas the accuracy of the measurements are influenced by the fission product and corrosion-coolant activation product concentrations in the primary coolant steam and water phases. Operational experience is required to establish the acceptable accuracy for the system at the various operating conditions.

That equipment which provides inputs to safety-related systems are designed to IEEE Standard 279-1971 as stated in 7.6.2.4.2.3.a and as such will have operability and calibration tests performed as stated in 7.6.2.4.2.3.c.

Miscellaneous revisions to 7.6.2.4 and 5.2.5 have been made in light of this question.

Q. 211.008
(5.2.2)
(7.6.1)

In Section 7.6.1.4 of the FSAR, you state that major components within the drywell which are sources of leakage by nature of their design (e.g., the sump seals, the valve stem packing, and the equipment warming drains), are enclosed and the leakage is piped to an equipment drain sump and identified there. Indicate what you mean by the term "sump seals" (i.e., did you intend to state pump seals). Discuss what monitors are available to the operator to permit him to identify the source of leakage; i.e., whether the leakage is from the "sump" (or pump) seals or the equipment warming drains or from any other component leakage sources which drain into the drywell equipment drain tank. Indicate whether there are any sumps within the drywell which must be filled before the sump drain flow is routed to the equipment drain tank.

Response:

The term "sump seals" is an error and should read "pump seals".

Sources of leakage which are enclosed and piped to the Drywell Equipment Drain Sump are monitored by the following means to provide the operator with information to permit him to identify the source of leakage.

a. Recirculation pump seals are provided with flow switches and control room alarms to identify excessive seal leakage.

b. The area between the vessel head seals is monitored with a pressure switch and a control room alarm to indicate inner seal leakage.

c. Remote operated valves within the Drywell are provided with seal leakoff lines. These lines connect between the valve stem packing are equipped with thermocouples to detect leakage through the inner packing and alarm in the control room.

All other sources of leakage which are piped to the drywell equipment drain tank are not considered as leakage paths. These lines are isolated by double block valves, which are opened only when needed.

Both the drywell equipment drain and floor drain sumps must fill to a specified level before reaching an overflow which routes liquid outside the drywell. See the response also to Question 211.002.

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Q. 211.009
(5.2.5)
(7.6.2)

In conformance with the staff's positions in Regulatory Guide 1.45, you state that the positions will be made to monitor systems connected to the reactor coolant pressure boundary for signs of intersystem leakage. Provide a detailed discussion of these provisions, including an identification of all potential intersystem leakage paths; e.g., the leakage from the primary coolant system to the residual heat removal (RHR) system and the emergency core cooling system (ECCS) injection line. Identify the instrumentation used in each path which will provide positive indication of intersystem leakage in the affected system.

Response:

Below is a listing of equipment which alert the operator to intersystem leakage from the reactor coolant pressure boundary into a low pressure system.

Each low pressure system process line is provided with an overpressure sensing pressure switch. The pressure switch activates an alarm in the control room to alert the operator to a degrading situation where a leaking shutoff valve may result in system pressure exceeding design limits.

The following is a list of process line, line design limits, associated overpressure sensor, and sensor setpoint activating an alarm.

Intersystem Leakage Path	Design Pressure	Sensor ID No.	Sensor Setpoint
RHR/Recirc Suction	220 psig	PS E12-N018	190 psig
RHR/Recirc Discharge	500 psig	PIS E12-N022A,B	400 psig
RHR Head Spray	500 psig	PIS E12-N022B	400 psig
LPCS Injection	550 psig	PS E21-N005	450 psig
LPCI Injection	500 psig	PIS E12-N022A,B,C	400 psig

Q. 211.010
(5.2.5)

Operating experience at some boiling water reactors (BWRs) has shown that the high pressure coolant injection system (HPCI) and the reactor core isolation cooling system (RCIC) have been rendered inoperable due to inadvertent leak detection isolations which were caused by a high differential temperature signal from the equipment room area. These isolations occurred when there was a relatively sharp drop in the outside temperature. In 5.4.6.1.1.1 and Table 5.2-9 of the FSAR you indicate that the WNP-2 facility also has this type of isolation for the RCIC system and the steam side of the RHR system. Provide a discussion of the modifications that have been, or will be, made to prevent inadvertent isolations of this type which effect the availability and reliability of the RCIC and the RHR systems. Additionally, indicate the trip settings you propose for isolation of the WNP-2 RHR and RCIC systems due to high area temperature in terms of degree Fahrenheit above the ambient temperature. Discuss the method you propose to avoid this problem. Show that the differential temperature setting could not be set too low, thereby causing an inadvertent isolation when the RCIC and the RHR systems are needed.

Response:

The RHR and RCIC pump rooms are located approximately 20 feet below grade level and would not be directly affected by a sharp drop in the outside air temperature. These rooms are ventilated by the reactor building ventilation system which contains a freeze protection device that will annunciate in the main control room if 40°F air existed downstream of the heating coil.

The high dfferential temperataure setting for both RHR and RCIC pump rooms is 70°F. The temperature elements are located at the face of the supply and exhaust ductwork in each room. The maximum room temperature rise could be 58°F, in one of the three RHR pump rooms, with RHR pump operating. It is, therefore, very unlikely that an inadvertent isolation of the RHR or RCIC systems could occur.

The high temperature trip setting for RHR and RCIC pump rooms is 150°F higher than the maximum calculated room temperature.

If any of the trip settings are too low, the consequences have no safety significance, because RCIC and the steam supply to the RHR heat exchanger are not required safety systems. The trip setting will be adjusted if necessary to reduce any operational inconveniences.

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Q. 211.018
(4.6.1)

Experience at some operating BWRs indicates that failures can occur in the collet fingers of the CRD mechanism. In order to resolve this problem, some BWR facilities under construction have installed a revised collet retainer design. However, you do not address this particular problem in your FSAR nor do you discuss its resolution. Accordingly, confirm whether the revised collet retainer design will be incorporated into the CRD mechanisms of the WNP-2 facility. Revise Table 1.3-8 of the FSAR as required.

Response:

There have been no failures of collet fingers reported from the field, and no design change is contemplated. General Electric has demonstrated by testing and operating experience that the existing CRDs meet all safety and licensing requirements and are expected to give full life performances. However, as a result of examining operating drives, General Electric has discovered evidence of Intergranular Stress Corrosion Cracking (IGSCC) in some CRD drive components.

To preclude the potential for increased maintenance, we plan to incorporate the latest product improvements as provided by GE during our routine maintenance of the CRDs. Along with the other parts of the drive, the collet retainer tube, piston tube, and index tube will be routinely checked and changed out, if necessary, considering the latest design improvement available at the time.

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Q. 211.032
(5.4.7)

Discuss the system design provisions incorporated into the facility to prevent damage to the RHR pumps in the LPCI mode during pump runout conditions in the ECCS operating mode and in the test mode. We note that Figures 5.4-13a, 5.4-13b and 5.4-14a of the FSAR indicate that a metering orifice is installed in the discharge lines. Indicate whether this metering orifice can perform the same function as a restricting orifice. If not, it is our position that the discharge lines of the RHR pumps should incorporate a restricting orifice.

Response:

The metering orifice in the discharge line does not serve as a restricting orifice.

The piping for each mode of RHR operation is investigated to ensure that the resistance is low enough to allow the rated flows given in Figure 5.4-14b yet high enough to prevent pump runout. Restricting orifices are necessary in the system test lines (for suppression pool cooling and test modes). Engineering changes are currently being processed which will add these restricting orifices. Restricting orifices in the LPCI lines are currently being investigated. The balance of the RHR modes do not require restricting orifices in their lines.

Q. 211.037
(5.4.7)

In Note 12 of Figure 5.14-13a of the FSAR, you state that: "Between valves MOF008 and MOF009 consideration should be given to thermal expansion of the contained water." Provide a commitment to incorporate a method for pressure relief between these two RHR isolation valves. Alternatively, show by analysis that piping integrity would be maintained in the event that a LOCA or stream line break occurred and the water trapped between these two valves, thermally expanded.

Response:

A pressure relief valve will be installed between MOF008 and MOF009. It will meet the following design requirements:

ASME III, Class 1	
Seismic Category 1	
Quality Class 1	
Relief Capacity	10 gpm
Design Pressure	1250 psig
Design Temperature	575°F
Set Pressure	150 psig
Size	3/4" x 1" flanged

The discharge of the relief valve will be back to the equipment drains.

Q. 211.038
(5.4.7)
(14.2.12)

Provide the test acceptance criteria discussed in Section 14.2.12.1.7 of the FSAR regarding preoperational testing of the RHR system.

Response:

Tentative acceptance criteria for the RHR preoperational test is attached.

Q. 211.087
(15.1)

It is not evident to us that the drop of 100° Fahrenheit which you assume in the feedwater temperature results in a conservative evaluation of the cold feedwater transient when the recirculation flow is manually controlled. For example, a feedwater temperature drop of about 150° Fahrenheit occurred at an operating BWR in this country as a result of a single failure of an electrical component. The electrical equipment malfunction which was a breaktrip of a motor control center, caused a complete loss of all feedwater heating due to a total loss of extraction steam. Accordingly, submit: (1) a sufficiently detailed failure modes and effects analysis to demonstrate the conservatism of the 100° Fahrenheit feedwater temperature drop you assume considering the potential effects of any single electrical malfunction; or (2) calculations using a limiting feedwater temperature drop which clearly bounds current operating experience.

Further, reductions in feedwater temperature less than 100° Fahrenheit can occur which would represent more realistic (i.e., slower) changes in feedwater temperature with time. In particular, slow transients with the surface heat flux in equilibrium with the reactor power when the reactor scrams due to a feedwater temperature drop smaller than 100° Fahrenheit, could result in a larger change in the critical power ratio (CPR). Accordingly, evaluate the cold feedwater transient for all sequences of events that can cause a slow transient and demonstrate the conservatism of the values of the feedwater temperature drops, including the rate of change with respect to time, which you assume in your present transient analysis.

Response:

The GE feedwater heater system design specification to the A/E requires that the maximum temperature decrease which can be caused by bypassing feedwater heater(s) by any equipment single failure or operator error should be less than or equal to 100°F. This is the basis of the assumed drop of 100°F in feedwater temperature in the analysis. To verify proper design by the A/E, a review of the feedwater system will be performed during the start-up test program to determine the most limiting single failure or operator error in terms of impact on feedwater temperature drop. A test will then be performed which simulates such a failure or error to confirm plant response, MCPR transient behavior and feedwater temperature drop.

From the analysis with the assumed drop of 100°F in feedwater temperature, it shows that reactor scram due to high thermal power occurs during the transient. It is evident that transients resulting from feedwater temperature decreases greater than 100°F would also result in reactor scram due to high thermal power. Therefore, the transients are not more severe than the one shown in the FSAR. The conclusion that a greater than 100°F feedwater temperature reduction does not result in more severe transients is substantiated by an analysis performed on the LaSalle docket in the response to LaSalle Question 212.142. Due to similarity of design, the analysis is applicable to WNP-2. The analysis assumed a feedwater temperature drop of 150°F which bounds observed operating experience.

It should be pointd out that a steady state condition (i.e., the surface heat flux in equilibrium with the neutron flux at the occurrence of scram) is assumed in determining MCPR during the transient. Therefore, reduction in feedwater temperature less than 100°F will not result in a larger ΔCPR than that reported in the FSAR.

Q. 211.113
(5.2.2)

It would appear that improper setpoints would be a credible common mode failure which could result in degradation of the pressure relief systems. Show that adequate safety margin has been included in the overpressurization analysis to protect against a common mode failure of the safety/relief valves to open at the prescribed values.

Response:

The overpressure protection analysis has been performed with numerous conservative input values (low scram reactivity, high void reactivity, no relief valve actuation, high spring setpoint, etc.) and still shows a significant margin to the vessel code limit of 1375 psig. This margin is about 80 psi with a high flux scram (assuming failure of the direct position scram) and thus would allow for a significant deviation in the spring setpoint.

In addition, there is significant conservatism in the spring setpoint assumed in the analysis.

As stated in the response to Question 211.053, the overpressure analysis divides the valves into five groups such that each accounts for 20% of the total required capacity. The setpoints derived from the analysis are purposely higher than the actual values to account for initial setpoint errors and instrument setpoint drift. The initial pressure value indicated by the analysis is 1177. The actual setpoints will provide for six valve actuations by the time that pressure level is attained. That is 33% of capacity versus the 20% developed in the analysis.

Adherence to ASME Section XI will require setpoint verification on a continuing basis and provide additional setpoint credibility.

Q. 211.115
(5.2.2)

Subsection 5.2.2.4.1 of the FSAR states that setpoints for the power actuated mode for each safety/relief valve are specified in Table 5.2-2. Table 5.2-2 provides a listing of the setpoints and valve capacities of the valves in the five safety mode groups (spring-operated mode), but no data are presented for the relief mode of operation. Provide the relief setpoint for each safety/relief valve in Table 5.2-2 and in Figure 5.2-6.

Response:

Table 5.2-2 has been updated to include the relief (power actuated) setpoints. This table has also been updated to show the increased lowest spring setpoint and corresponding capacity. The P&ID Data, Figure 5.2-6, has been updated to reflect this increased lowest spring setpoint and the correct valve arrangement. Relief setpoints will not be added to the updated Figure 5.2-6. With the relief setpoints on Table 5.2-2, one can relate the relief setpoints to the corresponding valve identification and spring setpoints found in Figure 5.2-6.

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Q. 211.124
(5.2.5)

It is unclear in subsection 5.2.5.2f of the FSAR whether comparative "grab" samples of the continuously monitored containment atmosphere can and will be taken on a periodic basis. Resolve this ambiguity. If "grab" samples are not to be taken, justify the omission of these comparative data.

Response:

The drywell air sampling system has provisions for taking "grab" samples. "Grab" samples will be taken periodically, as necessary, to check monitor calibration and to determine airborne concentrations inside the drywell prior to personnel entry.

Q. 211.137
(4.6.2.3.1.2)

Identify the layout studies done to assure that no interference exists which will restrict the passage of control rods and the preoperational test(s) that are used to show acceptable performance.

Response:

The clearance study that was generically applied to all BWRs 4 and 5 "C" lattice plants with 0.100" channels was used in October 1975 (reference GE Drawing 767E667, Rev. 0).

During initial preoperational testing, an observer who is in direct communication with the control room will observe the operation of each individual control rod and verify that there is no binding or restriction to rod motion and will listen for any scraping or binding noises which may signify rod misalignment. In addition, the function of each CRD drive line will be measured as indicated by the differential pressure developed across the CRD piston during notch withdrawal. These differential pressure traces will be compared to reference traces to assure proper operation and the absence of abnormal friction.

Q. 211.149
(15.0)

Provide a realistic range and permitted operating band for the exposure dependent parameters in Tables 4.4-1 and 15.0-2. In Table 15.0-2, provide assurance that values of parameters selected yield the most conservative results.

Response:

None of the thermal and hydraulic design characteristics shown in Table 4.4-1 are exposure dependent. Instead, they reflect the rated power and flow limits which characterize the core design.

In Table 15.0-2, the only exposure dependent parameters are the doppler coefficient, the void coefficient, and the scram reactivity. If the parameter is assumed not to vary during exposure, a conservative value is assumed to bound the realistic range. While doppler and void reactivity effects impact transient performance, the scram reactivity dominates the transient response. To provide assurance that the transient evaluations yield the most conservative results, the evaluations are performed at core exposure conditions expected to occur with the worst scram reactivity characteristic. The minimum scram reactivity for projected operation in BWRs occurs at the end of cycle exposure point, when the control rods are completely withdrawn from the core at rated power/flow conditions.

The scram reactivity characteristic varies slightly with exposure, but is most strongly affected by the core power distribution and the associated control rod configuration prior to a scram. The scram reactivity used in the analysis shown in Figure 15.0-2 presents a conservative lower bound on the minimum scram reactivity for WNP-2, and also defines the minimum scram characteristic for permitted operation. In addition, the Plant Technical Specifications define surveillance requirements on control rod scram times to insure that the minimum required scram reactivity is provided throughout the plant lifetime.

The doppler coefficient varies slowly with exposure and is expected to be valued from -0.1433 to -0.2358 cents/°F during rated power operation in cycle 1. There is no defined operation band for this parameter. The void coefficient varies slightly with exposure and is expected to fall in the range of -6.32 to -9.50 cents/% (rated voids) in cycle 1. Except for requiring that the void coefficient is negative, there is no defined operation band for this parameter.

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Q. 211.151
(15.0)

The analysis of transients and accidents in Chapter 15.0 does not state which of the RPS time response delays in Table 7.2-5 is used in the REDY computer model (NEDO-10802). For each transient and accident in Chapter 15.0, specify which delay time in Table 7.2-5 is used in the analysis and why the specified delay time is conservative.

Response:

In all Chapter 15 events, the "maximum overall response time" of Table 7.2-5 is utilized for each scram encountered and reported in each event scenario. By utilizing the maximum overall response time for RPS sensors and logic, the overall scram time is maximized, thus conservatively minimizing the reactivity insertion effects on the event.

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Q. 211.163
(15.0)

For the majority of events analyzed in Chapter 15, the recirculation flow control mode (automatic or manual) assumed in the analysis is not specified. Our concern is that the mode selected may not result in the most severe margins on MCPR and peak vessel pressure.

a. Specify the recirculation flow control mode assumed for each event analyzed in Chapter 15.

b. Specify the change in MCPR and peak vessel pressure that results in these parameters for each event if the opposite recirculation flow control mode had been assumed in the analysis.

Response:

a. All of the major transient events are simulated with the manual recirculation flow control mode. The analysis evaluated with this mode of operation is more severe in transient results because of the following:

By using the manual flow control option in a hypothesized transient, the recirculation controller would lose its communication with the master controller. For example, the master flow controller would be unable to adjust the core flow demand downward in response to an up-power transient and thereby change recirculation flow to mitigate the power transient. In effect, recirculation flow would remain constant until an interlock such as RPT initiates a flow runback on pump trip. As a consequence, the resultant increase in core flow response time tends to worsen the simulated transient, making it a conservative estimate of plant response.

b. A representative comparison of the effect of automatic vs. manual recirculation flow control can be seen in Chapter 15 of the FSAR, Figures 15.1-1 and 15.1-2. At present the Supply System does not intend to operate with the recirculation system in automatic flow control (AFC) following completion of the Startup Test Program.

Q. 211.167
(15.1.3.3.3)

The depressurization rate has a proportional effect on the voiding action of the core. For the "pressure regulator failure-open" transient, the assumed depressurization rate results in a L8 trip. The results are not consistent with GESSAR 238-732 where a lower depressurization rate results in a trip from low turbine inlet pressure. Explain this discrepancy and provide justification that the assumed trip provides the most restrictive margins on MCPR and peak vessel pressure.

Response:

If the depressurization rate is not sufficient to cause level swell to reach L8, then turbine inlet pressure will drop below the isolation setpoint and cause isolation and scram. Since power is being depressed as the pressure decreases (due to additional voiding in the core), this transient is less severe when you assume a slower depressurization rate. Therefore, the assumed L8 trip provides the most restrictive margins on MCPR and peak vessel pressure. The results of this transient in the GESSAR II 238 document (22A7000) assume that water level swell will reach Level 8 and cause reactor scram which is consistent with the Supply System analysis.

Q. 211.179
(15.2.4.3.2)

The "closure of all MSIVs" transient (closure time 3 sec.) results in a position switch scram at 0.3 second and indirectly causes a scram trip of the main turbine and generator due to the decrease in pressure sensed by the main turbine. From Figure 15.2-5, it cannot be determined whether or not a turbine stop valve and turbine control valve scram occurs during the time interval that the MSIVs are closing from the full open position to the 90% scram position. Indicate in Table 15.2-5 the time at which the above indirect scram trips occur and the times at which the TSVs and TCVs become fully closed.

Response:

Following an MSIV full isolation, one of several automatic turbine protection devices may actuate if the operator does not take direct action to trip the turbines. Either the reverse power device, which senses directional power flow from/to the main generator, or the "anti-motoring" device, which senses "no load conditions" via turbine first stage pressure, would, most probably, trip the turbine when the steam supply diminishes. Experience indicates that upon steam interruption and stored energy present in the moisture separator/reheaters can supply the low pressure turbines with sufficient steam to maintain positive load on the generator for several seconds (10-20 seconds is not uncommon). This phenomenon results in a natural time delay in the actuation of the reverse power device. The anti-motoring device energizes a relay with a set of time delay closure contacts that provide input to the turbine trip logic when turbine first stage pressure decreases to 0. Coupling this feature with the fact that the MSIVs must travel well beyond 10% closed to affect steam throttle pressure and cause actuation of the "no load" device there is a time delay on the "anti-motoring" protective turbine trip. The result is that no turbine trip would occur in the 0.3 second time interval from start of MSIV full isolation to the generation of a reactor scram signal at an MSIV position of >10% closed. Since the MSIVs would be fully shut before a turbine trip would occur, the stop valve closure would have no impact on either the reactor peak pressure or the CPR transient. Specification of the times of which the TSVs and TCVs become fully closed in Table 15.2-5 is, therefore, unnecessary.

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Q. 211.181
(15.2.6.3.4)

For the "loss of AC power" transients, it is indicated that the trip of the feedwater turbines may occur earlier than simulated if the inertia of the condensate and booster pumps is not sufficient to maintain feedwater pump suction pressure above the low suction pressure trip setpoint. The simulation of this transient assumes sufficient inertia and, thus, the feedwater pumps are not tripped until the time that level reaches the high water level trip setpoint (L8). What quantitative effect on MCPR and peak vessel pressure would an earlier trip (insufficient inertia) of the feedwater turbines have?

Response:

Amendment 11 of September 1980 modified 15.2.6 in response to NRC Questions 211.096 and 211.097.

In the revised analysis, the feedwater pumps are not tripped on high water level, but on low pressure at the turbine due to MSIV closure. An earlier trip of the feedwater pump would produce lower peak pressure and less delta CPR due to void reactivity effects on the core (i.e., less subcooling means lower power).

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Q. 211.186
(15.4.4.2.3)

You state that, for the incident involving an abnormal startup of an idle recirculation pump, "Attempts to start the pump at higher power levels will result in a reactor scram on flux." Since such a transient may be more severe than the one presented, supply the analysis beginning at a higher power level.

Response:

The start of an idle recirculation pump from higher power levels is not more severe than the one presently analyzed because the flow-based APRM thermal power trip would terminate the transient before peak surface heat flux could reach more severe levels. The analysis presented in 15.4.4 is a bounding case for pump starts because it utilizes Technical Specification Limiting Conditions for Operation (LCOs), a single operator error (SOE), and an abnormally high initial power level. The LCO assumed is the active recirculation loop operating at 50% of rated drive flow. The SOE assumed is the pump start with the cold leg water temperature in the idle loop at 100°F (nearly 400 degrees below its LCO temperature for pump start, i.e., nominal active loop temperature at operating pressure, 545°F, minus maximum allowed loop differential, 50°F, equals required temperature, 495°F verses 100°F). The initial power level was established as high as possible without experiencing an APRM trip (the transient comes within 30% of the thermal power trip). In actual practice, the power level established for a pump start is determined from power generation considertions in order to provide prudent margin from reactor scram. Higher initial power levels would cause a trip and thus produce less severe transients. Additional analysis is therefore unnecessary.

Q. 211.207
(6.3)
(5.2.2)

Subsection 5.2.2.10 of the FSAR states that the manual and automatic actuation of the relief mode for each safety/relief valve is to be verified in preoperational testing. Subsection 6.3.4.2.2 of the FSAR states that each individual ADS valve is manually actuated prior to or following a refueling outage. The spring setpoint (safety mode) of each valve is to be checked during bench tests during refueling outages. On what schedule will safety/relief valves, other than the ADS valves, be manually operated in the relief mode to verify that the valve is operational? How many of the safety/relief valves will be removed during each refueling outage to receive preventive maintenance and be tested?

Response:

During the startup test program, all of the main steam safety/relief valves (SRVs) are tested for proper operation. These tests include a documentation review to assure that the valves were properly installed, properly handled during transportation, storage, and installation, and were properly maintained as to cleanliness prior to performance of any tests. In addition, the air accumulator capacity, SRV nameplate set pressure and capacity are compared with the system design documentation for compliance. Actual mechanical tests include an operability check of the SRV discharge line vacuum breakers, actuation of the individual SRVs by each remote manual switch (main control room and/or remote shutdown panel) to demonstrate full lift, smooth stroke, and opening time characteristics, actuation of each SRV in the relief mode by stimulating its pressure switch, and a demonstration that each SRV accumulator (automatic depressurization system, ADS, and/or normal) has sufficient capacity to operate the SRV air actuator as required by the system design documentation. Finally, the ADS logic is fully tested for proper performance. Note that only the air actuator is exercised during many of the startup tests. This minimizes valve wear and unnecessary maintenance.

During the power ascension phase of the startup test program, each SRV is manually actuated at approximately 250 psig reactor pressure to demonstrate valve operability. At approximately 25% power each SRV is actuated a second time to measure discharge capacity and to demonstrate that no blockage in the SRV discharge line exists.

At commercial turnover the scope of SRV testing is governed by ASME Boiler and Pressure Vessel Code Section XI, Article IWV and Plant Technical Specifications. This article specifies the rules and requirements for inservice testing to verify operational readiness of the SRVs. This code section will be applied to both ADS and non-ADS valves alike. Supplemental tests of the ADS valves each operating cycle are required by the Technical Specifications. Applying Section XI, the SRV test schedule (in part) would be as follows:

Time Period	Number of Valves Tested	Total Tested	Elapsed Time
Cycle 1	6	6	1.5 years
2	4	10	2.5
3	4	14	3.5
4	4	18	4.5
5	4	4	1.0
6	4	8	2.0
7	4	12	3.0
8	4	16	4.0
9	2	18	5.0

Note that Article IWV-3200 would be applied to those valves tested under IWV-3500 upon return to power operation.

This combination of startup test program, technical specification surveillance and inservice inspection testing satisifies industry standards for SRV operability demonstrations. As outlined in the response to NRC Question 211.051, the Supply System is participating in the BWR Owners' Group for TMI Concerns on SRV reliability and intends to consider the group's recommendations/improvements for application to our SRV program.

Q 331.3

Provide tabulations by nuclide, of the expected airborne concentrations of radioactivity in areas normally occupied by operating personnel including tabulations of the radioactivity which result from the following sources:

a. Removal of the reactor pressure vessel head and associated internals

b. relief valve venting; and

c. movement of spent fuel.

Describe the models and parameters which are used in calculating these concentrations.

Response:

Please refer to 12.2.

Tabulations, by radionuclide, of the expected airborne concentrations of radioactivity in areas normally occupied by operating personnel can be found in Tables 12.2-14, 12.2-15 and 12.2-16. The related mathematical model(s) are described in 12.2.2.2 and 12.2.2.3.3. With respect to the sources identified in the question:

a. Refer to 12.2.2.3.4 of the FSAR. Radioactivity contribution to normally occupied areas due to RPV head and internals removal is analytically insignificant. Head gases will be vented to the HVAC via the Reactor Building sump vent filter/charcoal filter train prior to RPV head removal. Similarly, reactor coolant can be processed through the Reactor Water Cleanup System until the radioactivity level in the collant is at the desired level. In addition, Reference (1) presents actual data from operating BWR's for normal operations including refueling. The data is felt to be conservative due to the design differences between WNP-2 and the older plants used in this report, particularly in terms of ALARA.

(1) "Sources of Radionuclide at Boiling Water Reactors", EPRI, NP-495, February, 1978.

b. See 12.2.2.3.3 of the FSAR. The discharge of all relief valves on systems containing radioactive fluid is controlled, via piping systems, to either the equipment or floor drain systems, another part of the system itself or the suppression pool.

With reference to the equipment or floor drain systems, the receiving point for relief valve discharges is in a radiation zone equivalent or higher than the equipment being relieved. For discharge back to the system, the same is true. All relief valves, which relieve pressure in the turbine main steam, exhaust directly to the condenser.

c. Please see 12.2.2.3.6 of the FSAR.

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(1) "Sources of Radionuclide at Boiling Water Reactors", EPRI, NP-495, February, 1978.

Q. 360.3

Expand the discussion in 2.5.2.2.2, including supporting data, to provide more complete tectonic bases for the proposed boundaries of the tectonic provinces identified in Figure 2.5-59 and in the referenced section of the FSAR. In particular, indicate why you believe there is sufficient justification to establish a tectonic boundary between the northern Cascades and the Columbia Basin province.

RESPONSE:

The text of 2.5.2.2.2 beginning on page 2.5-118, has been revised to incorporate the response to this question.

Q. 360.4

In the Weston Geophysical Research, Inc., report, "Qualitative Aeromagnetic Evaluation of Structures in the Columbia Plateau and adjacent Cascade Mountain Area," March 28, 1978, Figure 13 shows several north to northwest trending aeromagnetic linears in the vicinity of Badger Mountain and Jump Off Joe Anticline. However, the Weston report does not discuss the origin or interpretations of these particular linears. The north trending linear crossing the Columbia River at the junction with the Snake River has an apparent offset of the magnetic low defining the Rattlesnake Hills anomaly. Since these aeromagnetic linears trend toward the WNP-2 site, provide: (1) an interpretation of these features, including but not limited to the potential for their continuation to the north to near site area; and (2) a discussion of the fault parameters, if such an interpretation is proposed.

Response:

The concerns raised in this question relate to recent information which post-dates the information now before the staff. The reference letter proposes a meeting to update the staff with respect to this information. As stated in the letter, a generic report is scheduled for early fall 1979 which will place this information in perspective and respond to the concerns of this question.

Reference:

Letter, D. L. Renberger (WPPSS) to O. D. Parr (NRC), "Update of Geological Studies", dated April 27, 1979.

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Q. 421.007

Describe the qualifications for the positions of QA Director and Project QA Manager.

Response:

For the QA Director, qualifications include a BS degree in engineering, or a related field, and ten years broad experience in design, procurement, construction and operations in the nuclear industry. Directly related experience may be substituted for academic requirements where the candidate's record of performance clearly demonstrates that the candidate is able to staff the position without question. Requirements include knowledge of generally accepted policies and procedures as they relate to Quality Asurance Programs; knowledge of the NRC Quality Assurance criteria; knowledge of significance of licensing commitments such as those documented on the FSAR and PSAR; knowledge of relevant regulations and rulings developed by Federal and State agencies; knowledge and ability in the areas of planning, organization, measurement, decision making; must be capable of broad overall performance under multiple project conditions.

Minimum qualifications for the Project QA Manager include a BS degree in engineering, or a related field, and ten years experience in nuclear quality assurance or technically related activities. Directly related experience may be substituted for academic requirements where the candidate's record of performance clearly demonstrates that the candidate is able to staff the position without question. Requirements include knowledge of generally accepted policies and procedures as they relate to Quality Assurance programs; knowledge of the NRC Quality Assurance criteria; knowledge of the generic PSAR and FSAR requirements; knowledge of relevant regulations and rulings developed by Federal and State agencies. Must have demonstrated ability to effectively manage a team of multidiscpline QA personnel and accomplish work within established plans, budgets and schedules.

Q. 421.008.

Give a brief summary of WPPSS' corporate QA policies.

Response:

Supply System Quality Assurance is responsible for establishing Quality Assurance policy, goals, and objectives through the development and administration of the Supply System QA Program. This program is defined in the Supply System QA Program Manual developed by the Manager, Quality Engineering, and reviewed and approved by the Director, Quality Assurance and endorsed by the Managing Director.

Supply System QA personnel have the authority and responsibility to perform any actions necessary, including Stop Work Authority, to accomplish their mandate as delineated in the Quality Assurance Manual. This responsibility and authority is stated in a Management Statement signed by the Supply System Managing Director. The Management Statement appears in each Quality Assurance Manual. In matters of conflict regarding Quality Assurance policies or the Quality Assurance organization's authority to enforce them at the working level, the Director of Quality Assurance has direct access to all levels of upper management including the Managing Director for satisfactory resolution.

Q. 421.041

Section 17.1.3 - General Electric's current activities related to WNP-2 should be in accordance with NEDO-11209-04A, Revision 2, dated February 1, 1980. In addition, General Electric should update their commitment to the following Regulatory Guides or describe acceptable alternatives: 1.28-Rev. 2; 1.58-Rev. 1; 1.144-Rev. 1; and 1.146.

Response:

The General Electric Nuclear Energy Business Group (NEBG) provides a BWR Quality Assurance (QA) program as documented in NEDO-11209-04A, Revision 2, October 1980. This QA program is structured to comply with the NEBG commitments to the quality-related regulatory guides and has been approved by NRC in a letter dated September 29, 1980.

In this letter, NRC states that should regulatory criteria or regulations change such that their conclusions about the NEBG QA program become invalidated, they will notify GE. For this reason, the QA program provides for periodic modification and/or updating of NEBG commitments to comply with quality-related regulatory guides relating to the NEBG scope of supply. The NEBG commitments to comply with regulatory positions, or NRC approved alternate positions to quality-related regulatory guides are then incorporated into the BWR QA program documentation.

As approved by NRC, the NEBG commitment to Regulatory Guide 1.28 is to Revsion 0 (June 1972). Regulatory Guide 1.144 has not been committed to by NEBG. The next revision of NEDO-11209-04A, will include the NEBG commitments to Regulatory Guides 1.58, Revision 1, and Regulatory Guide 1.146; these commitments have been verbally approved by NRC.

Q. 421.042

Section 17.1.4.3 - Bechtel should update their commitment to the following Regulatory Guides or describe acceptable alternatives: 1.28-Rev. 2; 1.39-Rev. 2; 1.123-Rev. 1; and 1.144-Rev. 1.

Response:

Bechtel's Regulatory Guide positions meet project and Supply System commitments in Appendix C.3. Bechtel will address Regulatory Guide 1.144, Revision 1 in conjunction with the Supply System.

See response to Question 421.014.

2. Additional venting capability is required for those portions of each hot leg that cannot be vented through the reactor vessel head vent or pressurizer. It is impractical to vent each of the many thousands of tubes in a U-tube steam generator; however, the staff believes that a procedure can be developed that assures sufficient liquid or steam can enter the U-tube region so that decay heat can be effectively removed from the RCS. Such operating procedures should incorporate this consideration.

3. Venting of the pressurizer is required to assure its availability for system pressure and volume control. These are important considerations, especially during natural circulation.

WNP-2 Position

Since the purpose of the Reactor Coolant System (RCS) vents is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, this requirement is not applicable to WNP-2. The design of the WNP-2 Reactor Pressure Vessel (RPV) (as described in Chapter 4), precludes noncondensible gases from inhibiting natural circulation cooling of the core. The gases which may be generated from the core would collect in the reactor dome above the water which covers the core. Natural circulation through the core would continue unaffected by the noncondensible gases in the reactor vessel dome. Hence, venting of the reactor coolant system is not necessary to ensure continued natural circulation.

II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST-ACCIDENT OPERATIONS

Position

With assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Clarification

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. These actions should be taken pending conclusions resulting in the long term degraded core rulemaking, which may result in a need to consider additional sources.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

II.K.1.22 Proper Functioning of Heat Removal Systems

Position

Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., RCIC) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure by which this action is taken in a timely sense. (IE Bulletin 79-08).

Clarification

None

WNP-Position

WNP-2 letter G02-80-107 of May 23, 1980, responding to IE Bulletin 79-08, provided the following:

The auxiliary heat removal systems provided to remove decay heat from the reactor core and containment following loss of the feedwater systems are:

High Pressure Core Spray System (HPCS)
Reactor Core Isolation Cooling (RCIC) System
Low Pressure Core Spray System (LPCS)
RHR System - LPCI Mode
RHR System - Suppression Pool Spray Mode
RHR System - Suppression Pool Cooling Mode
Residual Heat Removal (RHR/Low Pressure Coolant Injection (LPCI) System

The description that follows details the operation of the systems needed to achieve initial core cooling followed by containment cooling and then followed by extended core cooling for long-term plant shutdown, assuming the reactor is scrammed and isolated from the main condenser.

Initial Core Cooling:

Following a loss of feedwatr and reactor scram, a low reactor water level signal (level 2) will automatically initiate main steam line isolation valve closure. At the same time this signal will put the HPCS and RCIC Systems into the reactor coolant makeup injection mode. These systems will continue to inject water into the vessel until a high water level signal (level 8) automatically trips RCIC and closes the HPCS injection valve. The HPCS pump remains running on minimum flow bypass.

Following a high reactor water level 8 trip, the HPCS injection valve will automaticaly reopen when reactor water level decreases to low water level 2. The RCIC System must be manually reset by the operator in the control room before it will automatically re-initiate after a high water level 8 trip.

The HPCS and RCIC Systems have redundant supplies of water. Normally they take suction from the condensate storage tank (CST). The HPCS and RCIC System suctions will automatically transfer from the CST to the suppression pool if the CST ater is depleted or the suppression pool water level increases to a high level.

The operator can manually initiate the HPCS and RCIC Systems from the control room before the level 2 automatic initiation level is reached. The operator has the option of manual control after automatic initiation. The operator can verify that these systems are delivering water to the reactor vessel by:

a. Verifying reactor water level increases when systems initiate.

b. Verifying systems flow using flow indicators in the control room.

c. Verifying system flow is to the reactor by checking control room position indication of motor-operated valves. This assures no diversion of system flow to other than the reactor.

Therefore, the HPCS and RCIC can maintain reactor water level at full reactor pressure and until pressure decreases to where low pressure systems such as the Low Pressure Core Spray (LPCS) or Low Pressure Coolant Injection (LPCI) can maintain water level.

Containment Cooling:

After reactor scram and isolation and establishment of satisfactory core cooling, the operator would start containment cooling. This mode of operation removes heat resulting from safety/relief valve (SRV) discharge to the suppression pool. This would be accomplished by placing the Residual Heat Removal (RHR) System in the containment/suppression pool cooling mode, or the suppression pool spray mode, i.e., RHR suction from and discharge to the suppression pool.

The Operator could verify proper operation of the RHR system containment cooling function from the control room by:

a. Verifying RHR and Service Water (SW) system flow usinng system control room flow indicators.

b. Verifying correct RHR and SW system flow paths using control room position indication of motor-operated valves.

c. On branch lines that could divert flow from the required flow paths, closing the motor-operated valves and noting the effect on RHR and SW flow rate.

Extended Core Cooling:

When the reactor has been depressurized, the RHR system can be placed in the long-term shutdown cooling mode. The operator manually terminates the containment coling mode of one of the RHR loops and places the loop in the shutdown cooling mode as follows:

a. Trip the RHR pump to be used for shutdown cooling,

b. Close associated motor-operated valve in the suppression pool suction and LPCI discharge line to the vessel,

c. Open shutdown cooling suction valves from and discharge valves to the reactor vessel, and

d. Restart the RHR Pump.

In this operating mode, the RHR system can cool the reactor to cold shutdown. Proper operation and flow paths in this mode can be verified by methods similar to those described for the containment cooling mode.

In conclusion, the WNP-2 plant design is fully adequate to meet the intent of the requirements of auxiliary heat removal when the main feedwater system is inoperable.

II.K.1.23 Reactor Vessel Level Instrumentation

Position

Describe all uses and types of vessel level indication for both automatic and manual initiation of safety systems. Describe other redundant instrumentation which the operator might have to give the same information regarding plant status. Instruct operators to utilize other available information to initiate safety systems (IE Bulletin 79-08).

Clarification

None

WNP-2 Position

NEDO-24708 describes the multiple water level instrumentation provided in the BWR control room for the operator. An outline of the specific indication for WNP-2 is provided in the following paragraphs, which fully meets the intent of the plant requirements and the NRC requirements.

Reactor vessel water level in the WNP-2 BWR is continuously monitored by eleven (11) indicators or recorders for normal, transient and accident conditions. In general, those monitors used to provide manual safety equipment initiation are arranged in a redundant array with two instruments, one in each of two independent electrical divisions. Thus, adequate information is provided to the operator for manual initiation of safety actions and for assurance of the vessel water level at all times.

Those sensors used to provide automatic safety equipment initiation are arranged in a four quadrant vessel tap configuration with the four sensors divided electrically between two divisions.

In addition, the operating procedurs will reflect the requirements for the operators to also rely upon the information provided by other plant parameter indications relating to vessel level.

The range of reactor vessel water level from below the top of the active fuel area up to the top of the vessel is covered by a combination of narrow and wide-range instruments. Level is indicated and/or recorded in the control room.

II.K.3.17 Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specification Changes

Position

Several components of the emergency core cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outage for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation) (NUREG-0737).

Clarification

The present technical specifications contain limits on allowable outage times for ECC systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECC systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with the quantification of historical unreliability due to test and maintenance outages, which will be used to determine if a need exists for cumulative outage requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken. Test and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicant for an operating license shall establish a plan to meet these requirements.

WNP-2 Position

The position statement by NRC required a report to be submitted by operating plants to identify actual ECCS outage

experience for the last five years of operation. The report requirement does not apply to WNP-2 since no operating experience is available. The Tech Specs for WNP-2 will be prepared and submitted for NRC review and approval based on currently acceptable ECCS outage times, by December 30, 1981.

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12.3-22a	14
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14.2-18 through 14.2-20	7
14.2-22 and 14.2-23	12
14.2-25 through 14.2-27	7
14.2-28 through 14.2-30	12
14.2-31	20
14.2-32	7
14.2-33	20
14.2-34 and 14.2-35	7
14.2-37	7
14.2-38	7
14.2-39	20
14.2-40	20
14.2-41	7
14.2-42	12
14.2-43 and 14.2-43a	12
14.2-43b	7
14.2-44 through 14.2-53	12
14.2-54	12
14.2-54a	20

CHAPTER 14 (Cont'd)

TEXT PAGES	AMENDMENT
14.2-55	20
14.2-56	12
14.2-57	13
14.2-58 through 14.2-61	12
14.2-62	13
14.2-63	13
14.2-64 and 14.2-64a	12
14.2-65 and 14.2-65a	12
14.2-66	12
14.2-67	12
14.2-68	20
14.2-69	12
14.2-70	12
14.2-70a	20
14.2-71	7
14.2-72	12
14.2-73	12
14.2-74	20
14.2-75	20
14.2-76 through 14.2-78	12
14.2-80	12
14.2-81	12
14.2-82	20
14.2-83	12
14.2-84	12
14.2-85 and 14.2-85a	20
14.2-86	12
14.2-87	12
14.2-88	20
14.2-89	20
14.2-90	12
14.2-91 through 14.2-93	20
14.2-94	12
14.2-95	20
14.2-96	7
14.2-98	12
14.2-99	20
14.2-101 through 14.2-104	20
14.2-105	12
14.2-107	12

CHAPTER 14 (Cont'd)

TEXT PAGES	AMENDMENT
14.2-109 and 14.2-109a	20
14.2-110	7
14.2-112	12
14.2-112a	7
14.2-113	7
14.2-114	20
14.2.115	13
14.2-116	12
14.2-116a	20
14.2-117	7
14.2-118	12
14.2-119 and 14.2-119a	20
14.2-120	12
14.2-121	7
14.2-122	12
14.2-123	13
14.2-124 and 14.2-124a	20
14.2-125	13
14.2-126	12
14.2-128	20
14.2-129	7
14.2-130	12
14.2-132	20
14.2-132a	13
14.2-133	7
14.2-134 and 14.2-134a	20
14.2-135	12
14.2-136 and 14.2-137	13
14.2-138	20
14.2-138a	13
14.2-139	20
14.2-140	13
14.2-141	7
14.2-142 and 14.2-142a	12
14.2-143 through 14.2-145	12
14.2-146	7
14.2-147	12
14.2-148	20
14.2-149	20
14.2-151 through 14.2-154	7
14.2-155 through 14.2-158	20
14.2-159 and 14.2-159a	20
14.2-160	20

CHAPTER 14 (Cont'd)

FIGURES	AMENDMENT
14.2-1 and 14.2-2	7
14.2-3	20
14.2-4	20
14.2-5	7
14.2-6	20

CHAPTER 15

TEXT PAGES	AMENDMENT
15-xxi	8
15-xxii and 15-xxiia	8
15-xxxvii	5
15.0-14	11
15.0-15	11
15.0-20	11
15.0-22	21
15.1-2	21
15.1-16	20
15.1-27	21
15.1-31	12
15.2-20	21
15.2-40	21
15.2-48 through 15.2-50	11
15.2-52 through 15.2-57	11
15.2-65	17
15.2-66	21
15.2-67	21
15.2-68	8
15.2-69	17
15.2-71	21
15.2-71a	8
15.2-72 through 15.2-76	8
15.2-80a through 15.2-80d	8
15.4-8	5
15.4-19	20
15.4-49	5
15.6-30	20
15.6-33	14
15.6-34	9
15.6-34a	11
15.6-37	17
15.6-39	10
15.6-40	9
15.6-41	10

CHAPTER 15 (Cont'd)

TEXT PAGES	AMENDMENT
15.6-42	9
15.6-44 through 15.6-47	10
15.6-48	9
15.8-3	12

FIGURES	
15.A.6-8	11
15.A.6-12 and 15.A.6-13	11
15.A.6-14a and 15.A.6-14b	11
15.A.6-15	11
15.A.6-20	11
15.A.6-22 through 15.A.6-31	11
15.A.6-38 through 15.A.6-40	11
15.A.6-42	11
15.A.6-43 through 15.A.6-45	11
15.A.6-51 through 15.A.6-53	11

CHAPTER 17

TEXT PAGES	
17-i through 7-v	17
17.1-1	17
17.1-2 and 17.1-2a	21
17.1-3	17
17.1-4	17
17.1-5	20
17.1-6	17
17.1-7	20
17.1-7a through 17.1-7e	20
17.1-8 and 17.1-8a	20
17.1-9	20
17.1-10 through 17.1-16	17
17.1-17 and 17.1-17a	20
17.1-18	17
17.1-19 and 17.1-19a	20
17.1-20	20
17.1-21	17

CHAPTER 17 (Cont'd)

APPENDIX A

APPENDIX B

APPENDIX C

TEXT PAGES

C-iii	14
C-v	14
C.2-3	4
C.2-16	4
C.2-18 through C.2-20	13
C.2-40	13
C.2-44 and C.2-45	21
C.2-46	13
C.2-55	13
C.2-63	13
C.2-65 and C.2-66	5
C.2-68	9
C.2-70	7
C.2-72	7
C.2-82	13
C.2-86	14
C.2-90	14
C.3-5	5
C.3-9	13
C.3-11	5
C.3-14	12
C.3-17	2
C.3-17a	2
C.3-18	13
C.3-22	5
C.3-23	8
C.3-25	8
C.3-32	8
C.3-33	8
C.3-34	8
C.3-35	11
C.3-39	13
C.3-41 and C.3-41a	21
C.3-45	5
C.3-47	8
C.3-49	13
C.3-49a	9
C.3-51	13
C.3-54	8
C.3-55 and C.3-56	13

APPENDIX C (Cont'd)

TEXT PAGES	AMENDMENT
C.3-57	21
C.3-59	7
C.3-60	14
C.3-61	14
C.3-67	8
C.3-70	9
C.3-74	17
C.3-75	20
C.3-76	11
C.3-77	14
C.3-81	4
C.3-82	8
C.3-84	13
C.3-86	14
C.3-91	13
C.3-92	9
C.3-102	8
C.3-107	8
C.3-111	8
C.3-112	14

APPENDIX D

TEXT PAGES	
Title Page	17

APPENDIX E

TEXT PAGES	
Title Page	10
E.1-1	7
E.2-1	14
E.3-1	8
E.3-3 through E.3-5	12
E.3-5a	3
E.3-9	3
E.3-10	5
E.7-1 through E.7-5	13

APPENDIX E (Cont'd)

TEXT PAGES	AMENDMENT
E.9-1	8
E.9-2	5
E.9-3 and E.9-3a	8
E.10-1	5
E.11-1	20
E.12-1	12
E.12-2	16
E.12-3	16

APPENDIX F

TEXT PAGES	
Title Page	7

APPENDIX G
(DAR)

TEXT PAGES	
Title Page	7
1.1-2	14 (Rev. 3)
2.1-4	14 (Rev. 3)
2.3-1	14 (Rev. 3)
2.3-2	14 (Rev. 3)
2.4-1	14 (Rev. 3)
3.1-3 and 3.1-3a	14 (Rev. 3)

APPENDIX H

TEXT PAGES	
H.O-vii and H.O-viia	12
H.1.2-1	12
H.1.2-4 and H.1.2-4a	21
H.2.1-4	10
H.2.3-5 and H.2.3-5a	12
H.4.3-1 and H.4.3-1a	12
H.A.O-i	10

FIGURES	AMENDMENT
H.A-1a through h	10

NRC QUESTIONS AND RESPONSES
(See Separate List)

NRC QUESTION/FSAR SECTION
CROSS REFERENCE
(See Separate List)

VALVES

- AIR VENT VALVE
- GLOBE (NORMALLY CLOSED SHOWN WITH LOCK)
- GATE (NORMALLY OPEN) (SEE NOTE 4)
- GATE (NORMALLY CLOSED) (SHOWN WITH LOCK)
- GLOBE (NORMALLY OPEN) (SEE NOTE 4)
- GLOBE, STOP CHECK
- CHECK
- EXCESS FLOW CHECK
- BLEEDER CHECK VALVE
- PLUG OR BALL
- NEEDLE
- SQUIB ACTUATED SHEAR PLUG
- THREE WAY VALVE (FIG. A) SHOWN TO INDICATE EITHER: 1-PORT STATUS (OPEN OR CLOSED) NOT DETERMINED OR 2-MODULATING 3 WAY CONTROL VALVE (SEE NOTE 3)
- THREE WAY VALVE (FIG. B) NEXT PORT OF CLOSURE NORMALLY CLOSED PORT NORMALLY OPEN PORT
- FOUR WAY
- ANGLE
- ANGLE, STOP CHECK
- GLAND LEAK-OFF
- GLAND SEAL WATER
- SAUNDER'S TYPE
- SAFETY OR RELIEF
- DIAPHRAGM OPERATED CONTROL OPENS ON AIR FAILURE
- DIAPHRAGM OPERATED CONTROL CLOSES ON AIR FAILURE
- (APPLIES TO ALL TYPES OF VALVES. USED TO DESIGNATE DIAPHRAGM ACTION.)
- REMOTE ELECT. SETTING RELIEF VALVE
- BUTTERFLY
- COMBINED BLEEDER CHECK & STOP VALVE (ISO. CHECK)
- FIRE HYDRANT
- HOSE VALVE
- SOLENOID VALVE W/ BUILT IN FLOW CONTROL
- REVERSE CURRENT VALVE

OPERATORS

- ELECTRIC HYDRAULIC OPERATED VALVE (OR DAMPER)
- (LIMIT SWITCH)
- MOTOR OPERATED
- (LIMIT SWITCHES AS REQ'D) HYDRAULIC OPERATED PISTON
- (LIMIT SWITCHES AS REQ'D) AIR OPERATED
- NE (SEE NOTE 1)
- SOLENOID
- MANUAL REMOTE
- DIAPHRAGM
- FLOAT
- MANUAL TRIP & RESET
- WITH MANUAL OVER-RIDE
- DIAPHRAGM VALVES
- QUICK OPENING
- SELF ACTUATED PRESSURE CONTROL VALVE
- SELF ACTUATED TEMPERATURE CONTROL VALVE
- THERMOSTATICALLY ACTUATED CONTROL VALVE

SYMBOLS

- DRIP PAN ELBOW
- FE OR FX — PITOT TUBE (ANNULAR) FLOW ELEMENT
- PIPING SPOOL PIECE (SHOWN FOR FLOW TEST SECTION)
- FURNISHED WITH ASSOCIATED EQUIPMENT
- EJECTOR
- INCREASER, REDUCER
- THREADED CAP
- QUICK DISCONNECT
- WELD CAP
- HOSE CONNECTION
- BLIND FLANGE
- "Y" TYPE STRAINER
- CONICAL STRAINER IN LINE
- TRAP
- FLEXIBLE CONNECTION
- EQUIPMENT DRAIN FUNNEL
- FILTER
- TEMPORARY STRAINER
- RUPTURE DISC
- EXPANSION JOINT
- BASKET STRAINER (DUPLEX)
- BASKET STRAINER (SIMPLEX)
- SPECTACLE FLANGE
- METERING ORIFICE PLATE
- FLOW NOZZLE
- SIGHT FLOW GLASS
- FLOW METER (POSITIVE DISPLACEMENT)
- IN-LINE FLOW INDICATOR, TRANSMITTER, ETC. (SEE CODE)
- RESTRICTING ORIFICE
- INSULATING FLANGE
- INSTRUMENT AIR SUPPLY (70-100 PSIG)
- INSTRUMENT AIR SUPPLY (REDUCED)
- FLOOR (AREA) DRAIN
- MULTIPLE ORIFICE
- SEAL WATER TRAP
- ELECTRONIC (POWER) SUPPLY
- SAMPLE CONNECTION
- CLAMP COUPLING (VICTAULIC)
- FLOW REGULATOR
- PRIMARY CONTAINMENT PENETRATION
- SECONDARY CONTAINMENT PENETRATION
- SPRAY HEADER OR RING

ELECTRICAL

- TEMPERATURE SCANNER INPUT
- ALARM ANNUNCIATOR ON LOCAL PANEL
- CONVERTER (MILLIVOLT/CURRENT)
- CONVERTER (PNEUMATIC/ELECTRIC)
- CONVERTER (ELECTRIC/PNEUMATIC)
- CONVERTER (RESISTANCE/CURRENT)
- MOTOR (ELECTRIC)
- ELECTRONIC TRIP UNIT IN MAIN CONTROL ROOM
- COMPUTER INPUT
- NOT FULLY OPEN ALARM - USED WHEN SERVICE IS NOT DEFINED (USE "L" FOR LOW AND "H" FOR HIGH)
- NOT FULLY CLOSED ALARM — USED WHEN SERVICE IS DEFINED. SEE NOTE 2.
- ALARM
- TRANSMITTER
- OPEN / CLOSED — PILOT LIGHTS
- MULTIPOINT RECORDER
- LOCAL
- REMOTE PANEL OR RACK MOUNTED
- MAIN CONTROL ROOM (PANEL OR RACK MOUNTED)
- TIME DELAY SWITCH (PANEL OR RACK MOUNTED)
- TEMPERATURE SELECTOR SWITCH
- LOW AMPLITUDE (LAS) SELECTOR
- HIGH AMPLITUDE (HAS) SELECTOR
- PROPELLER TYPE AGITATOR
- ELECTRIC WATER HEATER

INSTRUMENTATION CODING (SEE NOTE "2")

MEASURED VARIABLE (FIRST LETTER) \ INSTRUMENT FUNCTION		CONTROLLING (SECOND & THIRD LETTER): RECORDING	INDICATING	NON-INDICATING	CONTROL VALVE	SUMMER	FUNCTION GENERATOR	MEASURING (SECOND & THIRD LETTER): RECORDING	INDICATING	OBSERVATION GLASS	PRIMARY ELEMENT	TEST POINT	TRANSMITTER	INTEGRATOR	AMPLIFIER	SAMPLER	INDICATING SWITCH	NON-INDICATING SWITCH	ALARMS	DETECTOR
		-RC	-IC	-C	-CV	-Σ	-ƒ	-R	-I	-G	-E	-X	-T	-Q	Am	Sm	IS	-S	-A	
AIR	A	—	—	—	—	—	—	—	—	—	—	—	—	—	—	ASm	AS	—	—	—
CONDUCTIVITY	C	CRC	CIC	—	CCV	—	—	CR	CI	—	CE	CK	—	—	—	CSm	CIS	CS	CA	—
DENSITY	D	DRC	DIC	DC	DCV	—	—	DR	DI	—	—	DX	—	—	—	—	DIS	DS	DA	—
DIFF. PRESS.	dP	dPRC	dPIC	dPC	dPCV	—	—	dPR	dPI	—	—	—	dPT	—	—	—	dPIS	dPS	dPA	—
FLOW	F	FRC	FIC	FC	FCV	FΣ	Fƒ	FR	FI	FG	FE	FX	FT	FQ	—	—	FIS	FS	FA	—
MOISTURE	M	MRC	MIC	—	—	—	—	MR	MI	—	ME	—	—	—	—	—	—	—	—	—
NEUTRON FLUX	N	NRC	NIC	NC	—	—	—	NR	NI	—	NE	NX	NT	NQ	NAm	—	—	—	NA	—
HYDROGEN ION CONC.	pH	pHRC	pHIC	pHC	pHCV	—	—	pHR	pHI	—	pHE	pHX	—	—	pHAm	pHSm	—	—	—	—
LEVEL	L	LRC	LIC	LC	LCV	—	—	LR	LI	LG	—	LX	LT	—	—	—	LIS	LS	LA	—
OXYGEN	O_2	—	—	—	—	—	—	O_2R	O_2I	—	O_2E	—	—	—	—	—	—	—	—	—
PRESSURE	P	PRC	PIC	PC	PCV	—	—	PR	PI	—	—	PX	PT	—	—	—	PIS	PS	PA	—
POSITION	Po	—	—	—	—	—	—	PoR	PoI	—	—	—	PoT	—	—	—	—	PoS	PoA	—
RADIATION	R	—	—	—	—	—	—	RR	RI	—	RE	RX	RT	—	RAm	RSm	—	RS	RA	—
SPEED	S	SRC	SIC	SC	—	—	—	SR	SI	—	SE	—	—	—	—	—	—	—	—	—
TEMPERATURE	T	TRC	TIC	TC	TCV	—	—	TR	TI	—	TE	TX	TT	—	—	—	TIS	TS	TA	—
TIME	t	—	—	tC	—	—	—	—	tI	—	—	—	—	—	—	—	tIS	tS	—	—
VIBRATION	Vb	—	—	—	—	—	—	—	—	—	—	—	—	—	VbAm	—	—	VbS	VbA	—
WEIGHT FACTOR	Wf	WfRC	WfIC	—	WfCV	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—
SMOKE		—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	SMA	SMD
LIMIT																		LmS		—

HVAC SYMBOLS & ABBREVIATIONS

- OUTSIDE AIR INTAKE LOUVER
- DAMPER (PARALLEL)
- DAMPER (OPPOSED BLADE)
- MANUALLY OPERATED DAMPER
- AUTO. DIAPHRAGM OPERATED DAMPER
- AUTO. DIAPHRAGM OPERATED DAMPER WITH POSITIONER, MOTOR OPERATED DAMPER
- FAN SHOWN WITH INLET VANE CONTROL DAMPER
- PROPELLOR TYPE FAN
- INCLINED DROP IN RESPECT TO AIR FLOW
- INCLINED RISE IN RESPECT TO AIR FLOW
- DUCT SIZE (1ST NO. IS SIDE SHOWN) 12×20
- SUPPLY DUCT
- EXHAUST OR RETURN DUCT
- ROUGHING FILTER OR PREFILTER
- HI-EFFICIENCY FILTER
- SUPPLY OR RETURN AIR
- EXHAUST AIR
- UNDERCUT DOOR
- LEAKAGE AIR
- LOUVERED DOOR
- HUMIDIFIER
- CHARCOAL FILTER
- UNIT HEATER (ELECTRIC-STEAM OR HOT WATER)
- SQUARE ELBOW WITH TURNING VANE
- VANEAXIAL OR TUBEAXIAL FAN
- VANEAXIAL FAN WITH AUTOMATIC VARIABLE PITCH BLADES OR INLET VANES
- BACK DRAFT DAMPER

HVAC SYSTEM DESIGNATIONS

SYSTEM	CODE
CONTAINMENT RECIRC AIR	CRA
CHILLED WATER	CH
CONDENSER WATER	COW
EXHAUST AIR	EA
HEATING HOT WATER	HHW
MIXED AIR	MA
OUTSIDE AIR	OA
POTABLE WATER-HOT	PWH
POTABLE WATER-COLD	PWC
RETURN AIR	RA
REFRIGERANT	RE
SANITARY DRAINAGE-(PLUMBING)	PSD
SANITARY VENTS-(PLUMBING)	PSV
STANDBY GAS TREATMENT	SGT
CONTAINMENT SUPPLY PURGE	CSP
CONTAINMENT EXHAUST PURGE	CEP
SUMP VENT	SVT
DIFFUSER	DIF

MISC. ABBREVIATIONS

ACC	ACCUMULATOR
FC	INDICATES CLOSES ON AIR OR ELECT. FAILURE
FO	INDICATES OPENS ON AIR OR ELECT. FAILURE
LC	LOCKED CLOSED
LO	LOCKED OPEN
NC	NORMALLY CLOSED
NO	NORMALLY OPEN
RMC	REMOTE MANUAL CONTROL
RMS	REMOTE MANUAL SWITCH
HS	HAND SWITCH
FAI	FAIL AS IS
HCU	HYDRAULIC CONTROL UNIT
CRD	CONTROL ROD DRIVE
SQ RT	SQUARE ROOT EXTRACTOR
A/S	AIR SUPPLY
E/S	ELECT. POWER SUPPLY
MS	MANUAL SWITCH
H/A	HAND/AUTO STATION
S/S	SELECTOR SWITCH
B.D.	BACKDRAFT DAMPER
S.D.	SPLITTER DAMPER
V.D.	VOLUME DAMPER
T.R.	TOP REGISTER (VERTICAL ABOVE 6'-0")
T.G.	TOP GRILLE (VERTICAL ABOVE 6'-0")
B.R.	BOTTOM REGISTER (VERTICAL BELOW 6'-0")
B.G.	BOTTOM GRILLE (VERTICAL BELOW 6'-0")
C.R.	CEILING REGISTER (HORIZONTAL WITH OR W/O CEILING)
C.G.	CEILING GRILLE (HORIZONTAL WITH OR W/O CEILING)
C.D.	CEILING DIFFUSER (HORIZONTAL WITH OR W/O CEILING)
A.T.D.	AIR TURNING DEVICE
F.R.P.	FIBERGLASS REINFORCED POLYESTER DUCT

HVAC EQUIPMENT LEGEND

ITEM	DESIGNATION
AIR CONDITIONING UNIT	AC
AIR HANDLING UNIT	AH
AIR WASHER	AW
AUTOMATIC DAMPER	AD
CHARCOAL FILTER	CF
CHILLER	CR
COOLING COILS	CC
CONDENSATE OR CONDENSER UNIT	CU
DEAERATOR UNIT	DU
DUST COLLECTOR	DC
ELECTRIC HEATING COIL	EHC
FAN COIL UNIT	FC
FANS	FN
FILTER (HVAC)	FL
FILTER UNIT	FU
FIRE DAMPER	FD
FUME HOOD	FH
HEATING & VENTILATING UNIT	HV
HEATING COILS	HC
HIGH EFFICIENCY FILTER	HF
HUMIDIFIER	HU
MOISTURE SEPARATOR	MS
MULTIZONE UNIT	MZ
PORTABLE FILTER UNIT	PF
PUMP	P
REHEAT COIL	RHC
ROOF VENTILATOR	RVT
STEAM TRAP	T
UNIT HEATERS (WATER)	WUH
UNIT HEATERS (ELECTRIC)	EUH
UNIT HEATERS (STEAM)	SUH
WATER HAMMER ARRESTOR	WHA

MECHANICAL AND NUCLEAR EQUIPMENT LEGEND

ITEM	DESIGNATION
TANK	TK
HEAT EXCHANGER	HX
PUMP	P
DRYER	DY
VALVE	V
GAS & LIQUID ANALYZER	GLA
BOILER	BLR
EXPANSION JOINT	EJ
FILTER (PIPING)	F
LUBE OIL CONDITIONER	LOC
MECHANICAL TRANSPORTATION (CRANES)	CRA
ENGINE	ENG
EVAPORATOR	EV
COMPRESSOR	C
STEAM TRAP	T
RELIEF VALVE	RV
STRAINER	ST
AIR RECEIVER	AR
DRIVE TURBINE	DT
MIXER	MX
DEMINERALIZER	DM
EXHAUSTER	EX
VAPORIZER	VZ
SPECTACLE FLANGE	SF
AIR CONTROL VALVE	ACV
EMERGENCY SHOWER & EYE WASH	SEW
DEMISTER	DMS
HOSE STATION	HS
FIRE HYDRANT	HY
RUPTURE DISC	RD
EDUCTOR	ED
FILL CONNECTION	FCN
FLAME ARRESTOR	FA
SILENCER	ES
HOSE	HSE
QUICK DISCONNECT	QDC
RESIN TRAP	RST
RESTRICTING ORIFICE	RO
EJECTOR	EJC
TANK FILL CONNECTION	TFC
QUICK CONNECT CPLG.	QCC
AIR VENT VALVE	AV
SNUBBER	SNB
FLEXIBLE CONNECTION	FLX
DRIP PAN ELBOW	DPE
ANGLE STOP CHECK	ASTCH
GLOBE STOP CHECK	GSTCH
BUTTERFLY	BFLY
MANIFOLD	MA
LABORATORY EQUIPMENT	LE

PIPING AND SYSTEM DESIGNATIONS (SEE NOTE "5")

SYSTEM	CODE	G.E. SY. NO.
AIR REMOVAL (RADIOACTIVE)	AR	—
AUXILIARY STEAM	AS	—
BACKWASH AIR	BA	—
BLEED (EXTRACTION) STEAM	BS	—
BOILER BLOWDOWN	BD	—
BOILER CHEMICAL FEED	BCF	—
CARBON DIOXIDE	C	—
CHEMICAL FEED	CF	—
CHLORINATION	CL	—
CIRCULATING WATER	CW	—
CIRCULATING WATER BLOWDOWN	CBD	—
CONDENSER DRAINS & VENTS	CND	—
SERVICE AIR (STATION)	SA	—
CONDENSATE (AUXILIARY STEAM)	CO	—
CONDENSATE (NUCLEAR STEAM)	COND	—
CONDENSATE FILTER DEMIN. SYSTEM	CPR	—
CONTROL AIR SYSTEM	CAS	—
CONTROL ROD DRIVE (HYDRAULIC)	CRD	C12
DEMINERALIZED WATER	DW	—
DIESEL EXHAUST	DE	—
HYDRAULIC	HY	—
DIESEL OIL	DO	—
DIESEL STARTING AIR	DSA	—
DIESEL COOLING WATER	DCW	—
EQUIPMENT DRAINS (RADIOACTIVE)	EDR	G11
EXHAUST STEAM	ES	—
FILTERED WATER	FW	—
FIRE PROTECTION	FP	—
FLOOR DRAINS (NON RADIOACTIVE)	FD	—
FLOOR DRAINS (RADIOACTIVE)	FDR	G11
FUEL OIL (TO AUX BOILERS)	FO	—
FUEL POOL COOLING	FPC	G41
GLYCOL	GY	—
HEATER DRAINS	HD	—
HEATER VENTS	HV	—
HEATING STEAM	HS	—
HIGH PRESSURE CORE SPRAY	HPCS	E22
HYDROGEN	H	—
INSTRUMENTATION (PROCESS)	PI	—
LOW PRESSURE CORE SPRAY	LPCS	E21
MAIN STEAM	MS	B22
MAIN STEAM LEAKAGE CONTROL	MSLC	
MISC. DRAINS (NONRADIOACTIVE)	MD	—
MISC. WASTES (RADIOACTIVE)	MWR	G11
NITROGEN (INERTING)	N	—
OFF GAS	OG	N64
PROCESS SAMPLES	PS	—
PROCESS SAMPLING (RADIOACTIVE)	PSR	—
PROCESS VENTS (RADIOACTIVE)	PVR	—
PROCESS WASTE (RADIOACTIVE)	PWR	—
REACTOR BUILDING CLOSED COOLING	RCC	—
REACTOR CORE ISOLATION COOLING	RCIC	E51
REACTOR FEEDWATER	RFW	—
REACTOR RECIRCULATION	RRC	B35
REACTOR WATER CLEANUP	RWCU	G33
RESIDUAL HEAT REMOVAL	RHR	E12
ROOF DRAINS	RD	—
SEALING STEAM	SS	—
STANDBY SERVICE WATER	SW	—
STANDBY LIQUID CONTROL	SLC	C41
STANDBY GAS TREATMENT	SGT	—
SULFURIC ACID	SAT	—
TOWER MAKEUP WATER	TMU	—
TURBINE BUILDING SERVICE WATER	TSW	—
TURBINE OIL	TO	—
HEATING STEAM VENT	HSV	—
HEATING STEAM CONDENSATE	HCO	
CONTAINMENT MONITOR SYSTEM	CMS	
CONTAINMENT ATMOSPHERE CONTROL	CAC	
CONTAINMENT NITROGEN (INERTING)	CN	
CONTAINMENT PURGE EXHAUST	CEP	
CONTAINMENT PURGE SUPPLY	CSP	
CONTAINMENT INSTRUMENT AIR	CIA	
CONTAINMENT VACUUM BREAKER	CVB	
MISCELLANEOUS VENTS	MV	
MISCELLANEOUS WASTE	MW	
CULVERT PIPE SLEEVE	CPS	
LEAK DETECTION	LD	E31
GAS (BOTTLED)	GB	
REACTOR PROTECTION SYS.	RPS	C72

NOTES

1- ALL SOLENOID VALVES SHOWN IN DE-ENERGIZED POSITION. "NE" DENOTES SOLENOID IS NORMALLY ENERGIZED DURING PLANT OPERATION. "ND" DENOTES NORMALLY DE-ENERGIZED.

2- TYPICAL INSTRUMENT DESIGNATION

LEVEL SWITCH — ELECT. LINE — REMOTE HIGH/LOW LEVEL ALARM — INSTRUMENT NUMBER

3- WHERE THREE WAY VALVE OPERATIONS ARE NOT FINAL; FIG. A IS USED. WHERE VALVE OPERATION IS FINAL, FIG. B IS USED.

4- "NORMALLY OPEN" SYMBOL (UNSHADED) APPLIES TO VALVES OPEN AT ANY TIME DURING NORMAL PLANT START UP, OPERATION & SHUT DOWN.

5- FOR OPERATING, DESIGN CONDITIONS, AND COMPLETE PIPING DESIGNATION SEE MECHANICAL CONTRACT SCHEDULE SHEETS, AND SPECIFICATION.

6- TYPICAL HVAC EQUIPMENT DESIGNATION

TOA-HV-1
- HEATING & VENTILATING UNIT AND N° (EQUIPMENT)
- OUTSIDE AIR (SYSTEM)
- TURBINE GENERATOR BUILDING (LOCATION FOR H.V.A.C. ONLY)

7- TYPICAL HVAC, PLUMBING AND PIPING DESIGNATION

4"RD(1)-1
- PRESSURE CLASSIFICATION: 1- 150#, 2- 300#
- ROOF DRAIN (SYSTEM)
- PIPE SIZE

BUILDING LOCATION LEGEND

- A AUXILIARY BUILDINGS
- C PART OF OR WITHIN PRIMARY CONTAINMENT
- L OFFSITE LOCALE
- M MAKEUP PUMP HOUSE
- O OUTDOORS ON SITE
- P PUMP HOUSE
- R REACTOR BUILDING
- S SERVICE BUILDING
- T TURBINE GENERATOR BUILDING
- W RADWASTE/CONTROL BUILDING
- D DIESEL GENERATOR BUILDING

PIPING AND SYSTEM CODE GROUPS

CODE GROUP	APPLICABLE CODE
A	ASME BOILER AND PRESSURE VESSEL CODE, SECTION III CLASS 1
B	ASME BOILER AND PRESSURE VESSEL CODE, SECTION III CLASS 2
C	ASME BOILER AND PRESSURE VESSEL CODE, SECTION III CLASS 3
D	ANSI B31.1.0 CODE FOR PRESSURE PIPING
E	NSSS ENGINEERED INSTRUMENTS

QUALITY ASSURANCE CLASSES

REQUIREMENTS FOR QUALITY ASSURANCE CLASSES ARE SPECIFIED IN THE CONTRACT SPECIFICATION.

SEISMIC REQUIREMENTS

REQUIREMENTS FOR SEISMIC CLASSES ARE SPECIFIED IN THE CONTRACT SPECIFICATION.

LINES

- PROCESS MAIN LINE
- PROCESS LINE AUX. & INSTR. LINE
- INSTRUMENT OR SERVICE AIR
- INSTRUMENT & CONTROL ELECT. LEADS
- INSTRUMENT CAPILLARY TUBING
- LINES DESIGNATED ON OTHER DIAGRAMS
- TYPICAL SUPPLY BREAK POINT (UNLESS STATED OTHERWISE, HEAVY PORTION BY PIPING CONTRACTOR) — BY DESIGNATED MFR.

AMENDMENT NO. 11
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WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

BURNS AND ROE PIPING AND INSTRUMENTATION DRAWING SYMBOLS

FIGURE 1.2-16

LIST OF FIGURES (Continued)

LIST OF FIGURES

TABLE 1.8-2 (Continued)

GE DRAWING NO.	FSAR FIGURE NO.	FSAR FIGURE TITLE
761E221AD	7.4-2a-e	RCIC (FCD)
732E187	7.4-3	Standby Liquid Control System (P&ID)
761E218	7.4-4	Standby Liquid Control System (FCD)
732E191AD	7.6-1a,b	Leak Detection System
732E118A	7.6-4	Neutron Monitoring System (IED)
761E596C	7.6-6a-g	Neutron Monitoring System (FCD)
761E592	7.7-2a,b	Control Rod Drive Hydraulic System (P&ID)
761E354	7.7-3a-h	Control Rod Drive Hydraulic System (FCD)
761E918	11.3-1	Off-Gas System - Low Temperature
761E908AD	11.3-2	Off-Gas System (P&ID)

TABLE 1.8-2 (Continued)

GE DRAWING NO.	FSAR FIGURE NO.	FSAR FIGURE TITLE
761E549	5.4-18	Filter/Demineralization System (P&ID)
731E931AD	6.3-1	HPCS (P&ID)
731E932AD	6.3-2	HPCS (Process Diagram)
921E868AD	6.3-5	LPCS (P&ID)
761E220AD	6.3-6	LPCS (Process Diagram)
732E170AD	7.2-1a-d	Reactor Protection System (IED)
761E372	7.2-8a,b,c 7.3-16a,b,c	Process Radiation Monitoring System (IED)
761E423	7.3-1	Reactor Water Clean-up System (FCD)
828E156AD	7.3-4a,b,c 8.3-24a,b,c	HPCS Power Supply System
731E931AD	7.3-7	HPCS (P&ID)
731E950AD	7.3-8a,b,c	HPCS System (FCD)
732E103	7.3-9a,b,c	Nuclear Boiler System (P&ID)
731E788	7.3-10a-f	Nuclear Boiler System (FCD)
921D868AD	7.3-11	LPCS (P&ID)
731E760	7.3-12a,b	LPCS (FCD)
731E961AD	7.3-13a,b	RHR (P&ID)
731E999	7.3-14a-e	RHR (FCD)
732E151AD	7.4-1a,b	RCIC (P&ID)

TABLE 1.8-2

CROSS REFERENCE - GE PIPING AND INSTRUMENTATION DRAWINGS

GE DRAWING NO.	FSAR FIGURE NO.	FSAR FIGURE TITLE
921D280	1.2-15	General Electric Piping and Instrumentation Drawing Symbols
209A4756	1.2-17	Logic Symbols for NSSS Functional Control Diagrams
761E712	3.2-1	Group Classification Diagram
761E952	4.6-5a,b	Control Rod Drive Hydraulic System (P&ID)
112D1448	4.6-6a	Control Rod Drive System (Process Diagram)
112D1448AD	4.6-6b-e	Control Rod Drive System (Process Diagram)
732E103	5.1-3a,b,c	Nuclear Boiler System (P&ID)
761E445	5.2-6	Nuclear Boiler System (P&ID Data)
761E289AD	5.4-2a,b,c	Recirculation System (P&ID)
732E151AD	5.4-9a,b	RCIC (P&ID)
761E205AD	5.4-10	RCIC (Process Diagram)
731E961AD	5.4-13a,b	RHR (P&ID)
731E966	5.4-14a	RHR (Process Diagram)
731E966AD	5.4-14b,c	RHR (Process Diagram)
732E129AD	5.4-16	Reactor Water Cleanup (P&ID)
112D1777	5.4-17a,b	Reactor Water Clean-up System (Process Diagram)

TABLE 1.8-1 (Continued)

BURNS & ROE DRAWING NO.	FSAR FIGURE NUMBER	FSAR DRAWING TITLE
M546	9.4-6	Heating & Ventilating-TG Bldg.
M547	9.4-11	HVAC-Service Bldg.
M548	3.2-19	HVAC Control Room & Critical Switchgear
M549	9.4-3	HVAC-Radwaste Bldg.
M550	9.4-4	HVAC-Chilled Water System
M551	3.2-20 9.4-7	HVAC-CW,SW & MUW Pump Houses and DG Bldgs.
M552	9.4-9	HVAC-LABS & Office Area
M553	9.4-10	HW Heating and Chilled Water System
M554	3.2-17	Primary Containment Atmospheric Control System
M555	9.4-5	HVAC-Off-Gas Charcoal Absorber Vault
M556	3.2-21	Containment Instrument
	9.3-2	Air System
M557	3.2-25	Main Steam Isolation Valve Leakage Control System

TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 8	8A	January 1981
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 9	7	December 1980
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 10	2	September 1980
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 11	5	February 1981

TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 765	Radwaste & Control Bldg. El. 501'-0" & 507'-0" Instr. & Cont. Conduit & Sheet 3	11	March 1981
E 765	Radwaste & Control Bldg. El. 501'-0" & 507'-0" Instr. & Cont. Conduit & Sheet 4	4	October 1980
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 1	21	July 1980
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 2	22	February 1981
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 3	30	November 1980
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 4	27	October 1980
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 5	29	February 1981
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 6	32	January 1981
E 766	Radwaste & Control Bldg. Cable Table Instr. & Cont. Conduit & Tray Plan Sheet 7	27	January 1981

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TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 519	Motor Valve & Misc. Control - Sheet 29	7	June 1980
E 519	Motor Valve & Misc. Control - Sheet 30	7	March 1980
E 519	Motor Valve & Misc. Control - Sheet 31	10	April 1981
E 519	Motor Valve & Misc. Control - Sheet 32	12	April 1981
E 519	Motor Valve & Misc. Control - Sheet 33	14	April 1981
E 519	Motor Valve & Misc. Control - Sheet 34	8	February 1981
E 519	Motor Valve & Misc. Control - Sheet 34A	2	April 1981
E 523	Switch Development	11	January 1981
E 733	Radwaste & Control Bldg. El. 501'-0" & 507'-0" Lighting Plan	8	November 1979
E 751	Radwaste & Control Bldg. El. 501'-0" & 507'-0" Power Conduit & Tray Plan	15	October 1980
E 765	Radwaste & Control Bldg. El. 501'-0" & 507'-0" Instr. & Cont. Conduit & Tray Plan Sheet 1	29	February 1981
E 765	Radwaste & Control Bldg. El. 501'-0" & 507'-0" Instr. & Cont. Conduit & Sheet 2	23	February 1981

TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 519	Motor Valve & Misc. Control - Sheet 15	14	April 1981
E 519	Motor Valve & Misc. Control - Sheet 16	7	August 1979
E 519	Motor Valve & Misc. Control - Sheet 17	10	August 1980
E 519	Motor Valve & Misc. Control - Sheet 18	8	May 1980
E 519	Motor Valve & Misc. Control - Sheet 19	12	July 1980
E 519	Motor Valve & Misc. Control - Sheet 20	8	January 1981
E 519	Motor Valve & Misc. Control - Sheet 21	8	June 1980
E 519	Motor Valve & Misc. Control - Sheet 22	8	January 1981
E 519	Motor Valve & Misc. Control - Sheet 23	5	October 1979
E 519	Motor Valve & Misc. Control - Sheet 24	14	April 1981
E 519	Motor Valve & Misc. Control - Sheet 25	7	February 1981
E 519	Motor Valve & Misc. Control - Sheet 25A	0	October 1980
E 519	Motor Valve & Misc. Control - Sheet 26	6	October 1980
E 519	Motor Valve & Misc. Control - Sheet 27	2	January 1980
E 519	Motor Valve & Misc. Control - Sheet 28	8	January 1981

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TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 519	Motor Valve & Misc. Control - Sheet 7	14	August 1980
E 519	Motor Valve & Misc. Control - Sheet 8	8	April 1981
E 519	Motor Valve & Misc. Control - Sheet 9	16	January 1981
E 519	Motor Valve & Misc. Control - Sheet 10	11	January 1981
E 519	Motor Valve & Misc. Control - Sheet 11	10	May 1980
E 519	Motor Valve & Misc. Control - Sheet 12	7	March 1980
E 519	Motor Valve & Misc. Control - Sheet 13	7	October 1980
E 519	Motor Valve & Misc. Control - Sheet 14	13	April 1981

TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 518	480V Switchgear - Sheet 1	3	October 1979
E 518	480V Switchgear - Sheet 2	7	May 1980
E 518	480V Switchgear - Sheet 3	5	October 1980
E 518	480V Switchgear - Sheet 4	6	February 1981
E 518	480V Switchgear - Sheet 5	5	October 1979
E 518	480V Switchgear - Sheet 6	9	April 1981
E 519	Motor Valve & Misc. Control - Sheet 1	4	October 1980
E 519	Motor Valve & Misc. Control - Sheet 1A	3	May 1980
E 519	Motor Valve & Misc. Control - Sheet 1B	12	April 1981
E 519	Motor Valve & Misc. Control - Sheet 1C	0	October 1980
E 519	Motor Valve & Misc. Control - Sheet 2	2	September 1979
E 519	Motor Valve & Misc. Control - Sheet 3	8	April 1980
E 519	Motor Valve & Misc. Control - Sheet 4	10	January 1981
E 519	Motor Valve & Misc. Control - Sheet 5	9	April 1981
E 519	Motor Valve & Misc. Control - Sheet 6	8	January 1980

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TABLE 1.7-1(Continued)

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 517	4160V Switchgear - Sheet 9	11	December 1980
E 517	4160V Switchgear - Sheet 10	8	December 1980
E 517	4160V Switchgear - Sheet 11	8	December 1980
E 517	4160V Switchgear - Sheet 12	7	January 1981
E 517	4160V Switchgear - Sheet 13	5	December 1980
E 517	4160V Switchgear - Sheet 14	7	December 1980
E 517	4160V Switchgear - Sheet 15	8	December 1980
E 517	4160V Switchgear - Sheet 16	5	December 1980
E 517	4160V Switchgear - Sheet 17	6	December 1980

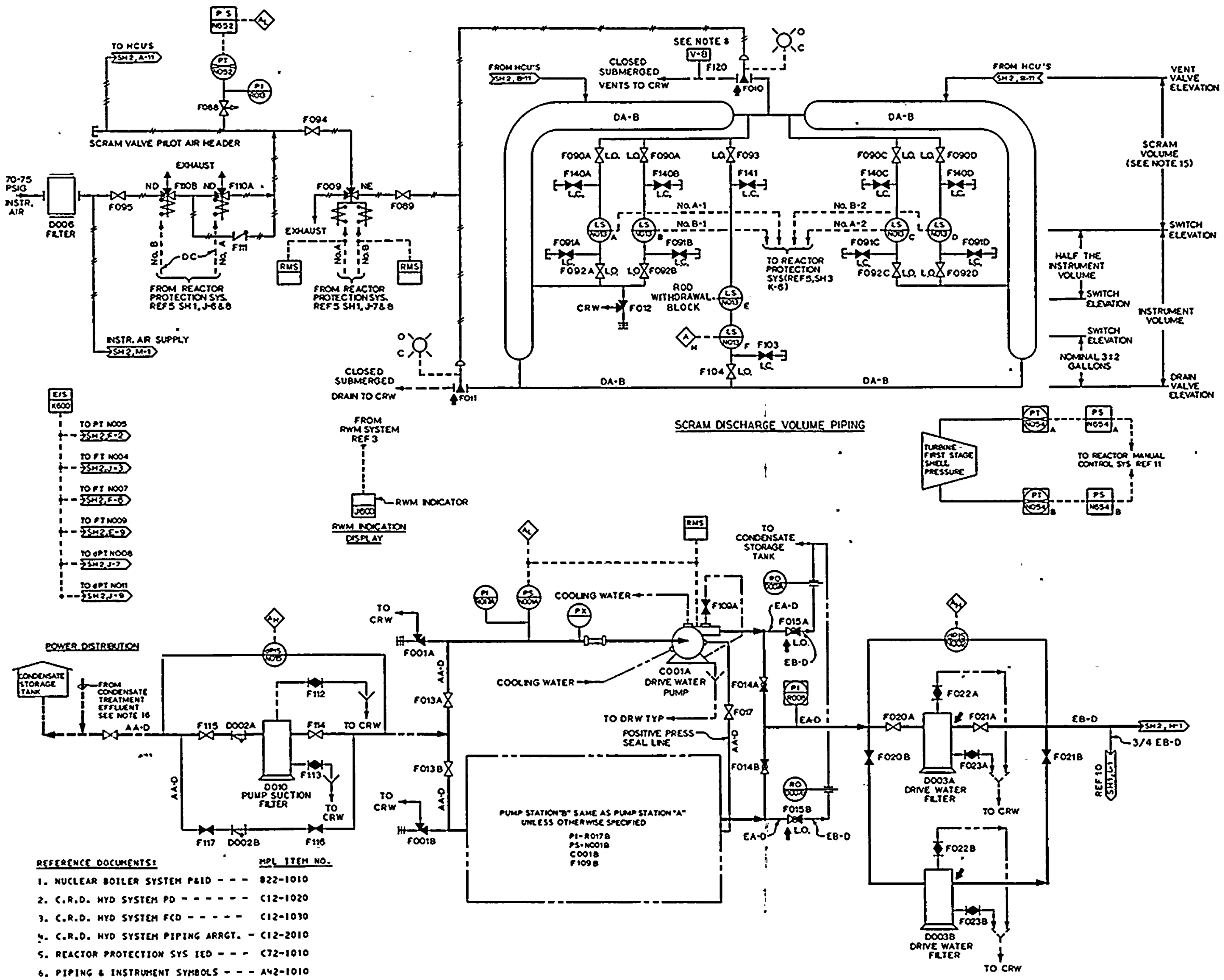
TABLE 1.7-1(Continued)

B. Burns and Roe, Inc. Drawings

Drawing No. (B&R Dwg.No.)	Title	Revision No.	Revision Submittal Date
E 501	Electrical Symbol List	8	January 1981
E 516	6900V Switchgear - Sheet 1	6	January 1981
E 516	6900V Switchgear - Sheet 2	7	January 1981
E 516	6900V Switchgear - Sheet 3	5	December 1980
E 516	6900V Switchgear - Sheet 4	4	December 1980
E 516	6900V Switchgear - Sheet 5	4	December 1980
E 517	4160V Switchgear - Sheet 1	5	December 1980
E 517	4160V Switchgear - Sheet 2	9	November 1980
E 517	4160V Switchgear - Sheet 3	9	November 1980
E 517	4160V Switchgear - Sheet 4	6	February 1980
E 517	4160V Switchgear - Sheet 5	8	January 1981
E 517	4160V Switchgear - Sheet 6	8	December 1980
E 517	4160V Switchgear - Sheet 7	9	April 1981
E 517	4160V Switchgear - Sheet 8	5	December 1980

TABLE 1.7-1 (Continued)

Drawing No. (G.E. MPL No.)	Title	Revision No.	Revision Submittal Date
G11-1040 807E174TC	Radwaste System Sheet 53	9	June 1979
G11-1040 807E174TC	Radwaste System Sheet 54	9	June 1979
G11-1040 807E174TC	Radwaste System Sheet 55	9	June 1979
G11-1040 807E174TC	Radwaste System Sheet 56	7	Original
G11-1040 807E174TC	Radwaste System Sheet 57	9	June 1979
G11-1040 807E174TC	Radwaste System Sheet 58	7	Original
G11-1040 807E174TC	Radwaste System Sheet 59	9	June 1979
G11-1040 807E174TC	Radwaste System Sheet 60	9	June 1979
G11-1040 807E174TC	Radwaste System Sheet 61	7	Original
G11-1040 807E174TC	Radwaste System Sheet 62	5	Original



1. VALVE F007A-A CLOSES ON DRIVE INSERT SIGNAL. VALVE F007A-B CLOSES ON DRIVE WITHDRAWAL SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.(SH 2,K-9.
2.
3. STABILIZING VALVE F007B IS AN ALTERNATE FOR STABILIZING VALVE F007A.(SH 2,L & M-9)
4. PROVIDE VENT VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM HIGH POINTS.
5. PROVIDE DRAIN VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM LOW POINTS.
6. PROVIDED FOR SYSTEM FLUSHING.(SH 2,K & L-8)
7. AVAILABLE FOR TEMPORARY CONNECTION FOR INSTRUMENT FLUSHING. NO PERMANENT PIPING CONNECTIONS TO BE MADE TO THIS VALVE. (SH 2, E-9)
8 VACUUM BREAKER TO BE ON HIGH POINT OF VENT LINE.
9. EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING. THE PIPING DESIGNER SHALL SIZE PIPES IN CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.
10. FOR LOCATION & IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (C12-3050).
11. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NUMBER (CJ2) UNLESS OTHERWISE NOTED.
12. CRD NITROGEN & AIR LINES SHALL BE OF A NON-CORRODING MATERIAL.
13. THESE VALVES MUST BE OPEN FOR DRIVE MOVEMENT.
14. MULTIPLE ORIFICES CONNECTED IN SERIES AS SHOWN IN PURCHASE PART DRAWING OF ORIFICE D008. THE PRESSURE DROP ACROSS EACH ORIFICE IS 250 PSI AT PUMP RUN-OUT CONDITION. SEE MPL FOR THE QUANTITIES OF ORIFICES. VALVE F034 SUPPLEMENTS THE ORIFICES D008 FOR THE REQUIRED PRESSURE DROP.
15. THE SCRAM DISCHARGE VOLUME ARRANGEMENT SHOWN IS FOR REFERENCE ONLY. SEE CRD DESIGN SPECIFICATION FOR REQUIREMENT.
16. SOURCE OF CRD SYSTEM WATER SHALL BE NORMALLY FROM CONDENSATE TREATMENT SYSTEM, CONDENSATE STORAGE TANK IS THE ALTERNATE SOURCE IF CONDENSATE TREATMENT SYSTEM IS NOT IN OPERATION. FOR DETAILED DESIGN REQUIREMENTS FOR SOURCE AND QUALITY OF WATER, SEE REF 9.
17. DELETED
18. PIPING QUALITY CLASS EXTENDS TO CONNECTIONS WITH HCU. HCU DIAGRAM IS SHOWN FOR INFORMATION ONLY. FOR QUALITY CLASS OF THE HCU SEE GROUP CLASSIFICATION DIAGRAM.
19. DELETED
20. EXHAUST WATER FROM ONE MOVING DRIVE IS DISPERSED VIA THE HYDRAULIC CONTROL UNITS OF OTHER-NON MOVING DRIVES TO THE REACTOR PRESSURE VESSEL.

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. NUCLEAR BOILER SYSTEM P&ID - - -	B22-1010
2. C.R.D. HYD SYSTEM PD - - - - - - -	C12-1020
3. C.R.D. HYD SYSTEM FCD - - - - - -	C12-1030
4. C.R.D. HYD SYSTEM PIPING ARRGT. -	C12-2010
5. REACTOR PROTECTION SYS IED - - -	C72-1010
6. PIPING & INSTRUMENT SYMBOLS - - -	A42-1010
7. PROCESS INSTRUMENT PIPING & TUBING SPEC. - - - - - - - - - - - -	A62-4070
8. PRESS. INTEGRITY OF PIPING & EQUIP. PRESS. PARTS - - - - - - - - -	A62-4030
9. C.R.D. HYD SYSTEM DESIGN SPEC - -	C12-4010
10. REACTOR RECIRC SYS. P&ID - - - -	B35-1010
11. REACTOR MANUAL CONTROL ELEM DIAG-	C11-1050
12. REACTOR WATER CLEAN-UP SYS P&ID -	G33-1010

LEGEND:

= SOLENOID OPERATED VALVE W/ BUILT-IN FLOW CONTROL

= BLOCK TYPE NEEDLE VENT VALVE (ANGLE PATTERN)

LDS = LEAKAGE DETECTOR SWITCH
RWM = ROD WORTH MINIMIZER

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID	FIGURE 4.6-5a

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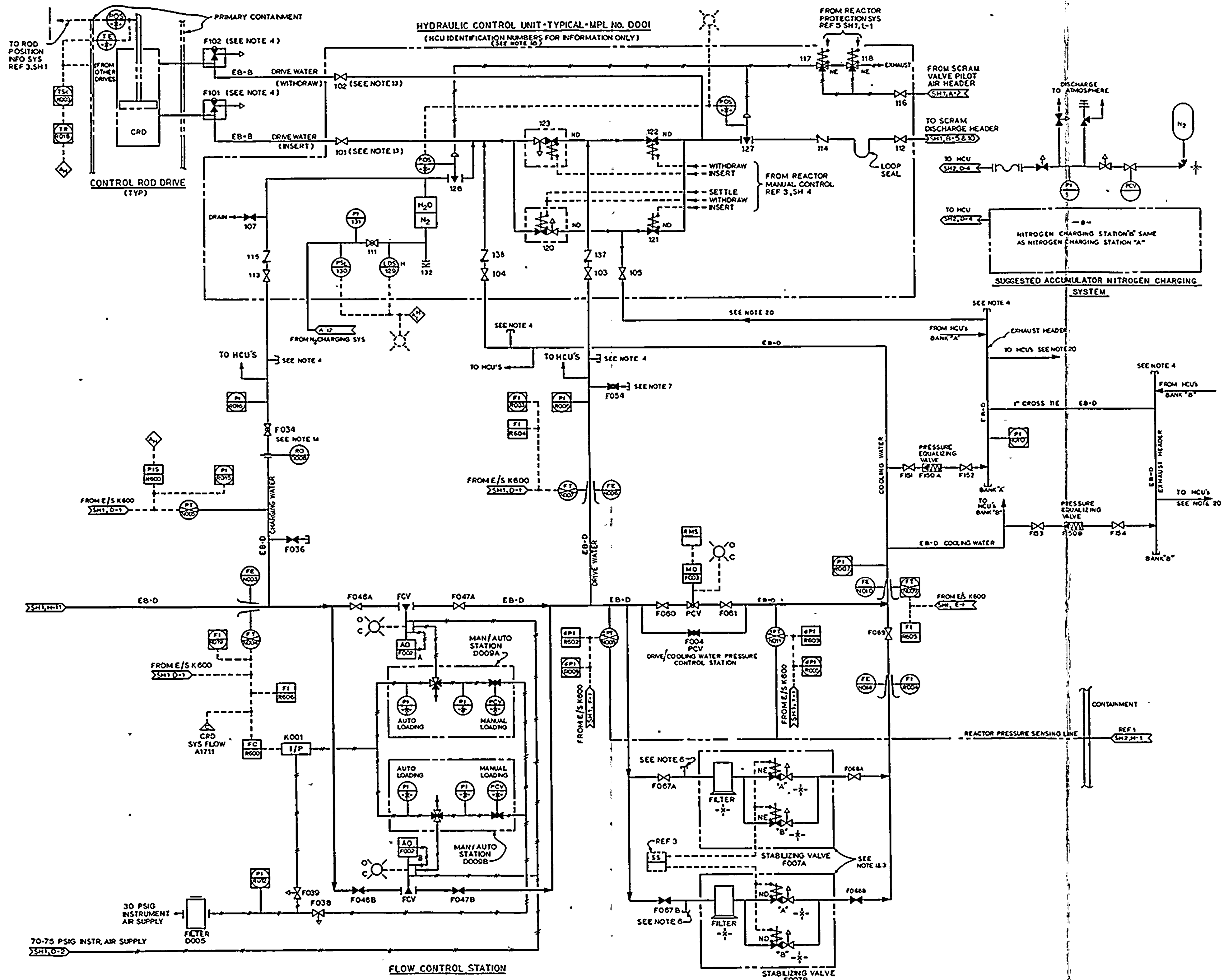

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE HYDRAULIC SYSTEM P&ID	FIGURE 4.6-5b

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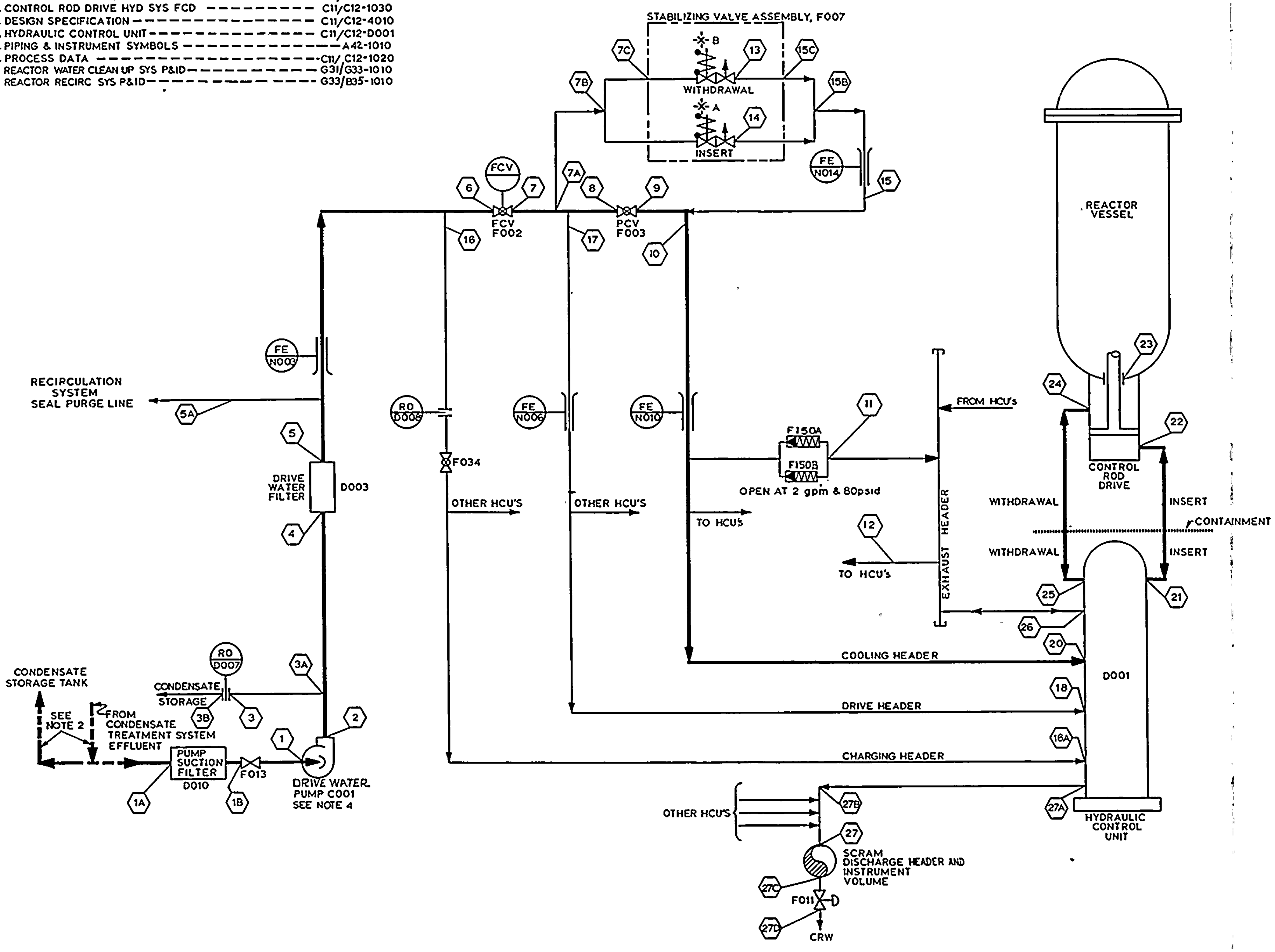
(C12-1020)

NOTE:

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NUMBERS.

REFERENCE DOCUMENTS:	MPL ITEM No.
1. CONTROL ROD DRIVE HYD SYS P&ID	C11/C12-1010
2. CONTROL ROD DRIVE HYD SYS FCD	C11/C12-1030
3. DESIGN SPECIFICATION	C11/C12-4010
4. HYDRAULIC CONTROL UNIT	C11/C12-D001
5. PIPING & INSTRUMENT SYMBOLS	A42-1010
6. PROCESS DATA	C11/C12-1020
7. REACTOR WATER CLEAN UP SYS P&ID	G31/G33-1010
8 REACTOR RECIRC SYS P&ID	G33/B35-1010



NOTES:

1. FOR DATA PERTAINING TO NUMBERS WITHIN HEXAGONS REFER TO PROCESS DATA REF 6.
2. SOURCE OF CRD SYSTEM WATER SHALL BE NORMALLY FROM CONDENSATE TREATMENT SYSTEM. CONDENSATE STORAGE TANK IS THE ALTERNATE SOURCE IF CONDENSATE TREATMENT SYSTEM IS NOT IN OPERATION. FOR DETAILED DESIGN REQUIREMENTS FOR SOURCE AND QUALITY OF WATER, SEE REF 3.
3. DELETED
4. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE SYSTEM (PROCESS DIAGRAM)	FIGURE 4.6-6a

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December 1981

MPL NO. C12-1020

(FOR NOTES SEE SHEET 2.)

MODE A NORMAL OPERATION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	63	0	0	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	PR	PR	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	4	6	0	0	0		.34 MAX	.34 MAX	.34 MAX	.34 MAX	0	0	0	0	
PRESSURE PSIG	PR + 30	PR + 30	1455				PR + 15	PR + 14	PR + 14	PR	PR		PR	0	

CONDITIONS:
1. DRIVES LATCHED
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM COOLING FLOW TO DRIVES. MINIMUM REQUIRED PRESSURE AT POSITION 1A IS 20 FEET OF WATER AT 200 GPM.

MODE A SIZES THE COOLING WATER HEADERS.

LINE LOSS FROM LOCATION 10 TO LOCATION 20 SHALL NOT EXCEED 3 PSIG.

MODE B ROD INSERTION

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	93	93	93	20	73	73	10	63	63	57	57	59	0	.7	2
PRESSURE PSIG	21	19	1487	1487	1476	1462	1462	1455	PR + 260	PR + 260	PR + 30	PR + 30	<PR + 8	<PR + 8	PR + 30

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	2	0	4	4		0	4	4	1.3	.7	.7	.7	0	
PRESSURE PSIG	PR + 30	PR + 30	1455	PR + 260	PR + 250		PR + 15	PR + 91	PR + 90	PR	PR + 20 MAX	PR + 20 MAX	PR + 8 MAX	0	

CONDITIONS:
1. DRIVE INSERTING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. MAXIMUM DRIVING FLOW TO DRIVES

MODE B SIZES THE DRIVE WATER HEADERS.

MODE C SCRAM

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	45	45	45	20	25	25	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	21	1550	1550									SEE NOTE 9	SEE NOTE 9	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	0	0	0		0	90	90	-3.6	30	30	0.1 + SEE NOTE 9	APPROX 5565	
PRESSURE PSIG								1167 MIN	731 MIN	PR	256 MAX	94		65 MAX	
														SEE NOTE 10	

CONDITIONS:
1. DRIVES SCRAMMING
2. PRESSURE OF REACTOR (PR) AT 1000 PSIG.
3. FLOWS BASED ON MAXIMUM ROD VELOCITY OF 85 INCHES PER SECOND.

MODE C SIZES THE INSERT AND WITHDRAW LINES.

MODE D SCRAM COMPLETED

LOCATION	1A	1	2	3	4	5	5A	6	7	8	9	10	11	12	13
FLOW, GPM	200	200	200	20	180	180	10	15	15	15	15	15	15	14.9	0
PRESSURE PSIG	21	19	1210						>PR	>PR	>PR	>PR	>PR	>PR	

LOCATION	14	15	16	17	18		20	21	22	23	24	25	26	27	
FLOW, GPM	0	0	155	0	0		0	0.92	0.92	0.92	SEE NOTE 9	SEE NOTE 9	0.1	0	
PRESSURE PSIG			988					76	76	PR	65 MAX	65 MAX		65 MAX	
											SEE NOTE 10				

CONDITIONS:
1. SCRAMMING OF DRIVES COMPLETED
2. PRESSURE OF REACTOR (PR) AT 0 PSIG.
3. MAXIMUM CRD SUPPLY PUMP FLOW.

MODE D SIZES THE PUMP SUCTION LINE.

NOTE: MINIMUM ACCUMULATOR PRECHARGE PRESSURE IS 565 PSIG.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTROL ROD DRIVE SYSTEM	FIGURE 4.6-6b

TABLE I

LOCATION	1A-1B	1B--1	2---6	3A-3B	6--9	7A-7B	7B-7C
DESIGN PRESS. (PSIG.)	150	150	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	4	4	2	1	1.5	1	0.75
LOCATION	10-20	11-12	15B-15C	15-15B	16-16A	17-18	12-26
DESIGN PRESS. (PSIG.)	1750	1750	1750	1750	1750	1750	1750
DESIGN TEMP. (DEG F)	150	150	150	150	150	150	150
ESTIMATED LINE SIZE (INCHES)	2**	1	0.75	1	2	1	1
LOCATION	21-22	24-25	27A-27B	27B-27	27-27C	27C-27D	5A
DESIGN PRESS. (PSIG.)	1750	1750	1250	1250	1250	1250	1750
DESIGN TEMP. (DEG F)	150	550	280	280	280	280	150
ESTIMATED LINE SIZE (INCHES)	1	0.75	0.75	*	10	2	.75

* SEE CRD SYSTEM DESIGN SPECIFICATION.

** 2 INCH HEADER TO EACH HALF OF THE TOTAL QUANTITY OF HCU'S.

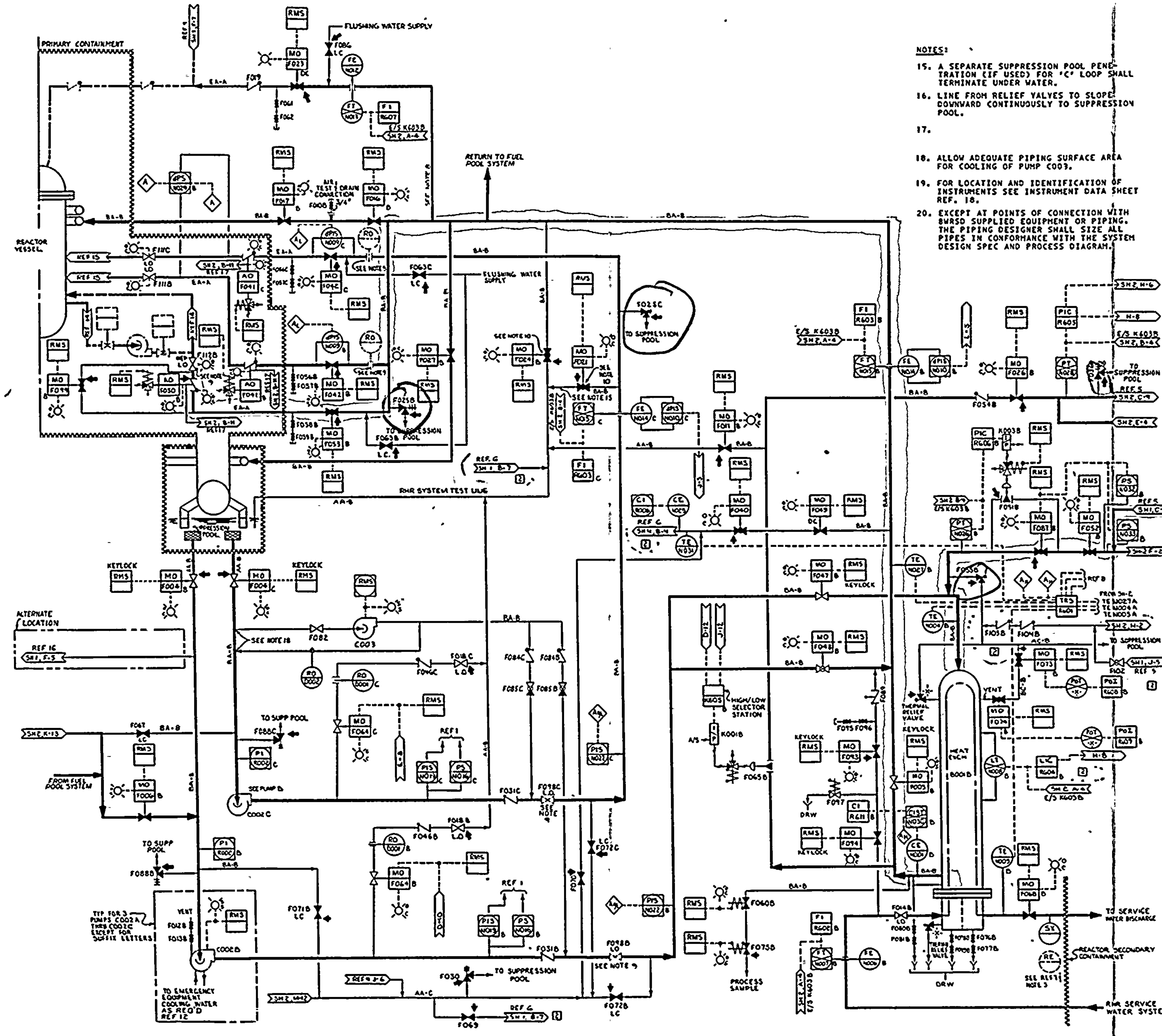
NOTES:

1. DEFINITION OF SYMBOLS
   PR - INDICATES PRESSURE OF THE REACTOR
2. MAXIMUM OPERATING TEMPERATURES
   THE MAXIMUM SYSTEM OPERATING TEMPERATURE WILL NOT EXCEED 150 DEG. F. FROM LOCATION 1 THROUGH 27 WITH THE FOLLOWING EXCEPTIONS.

	LOCATION	MAXIMUM TEMP. (DEG. F.)
MODE A -	23	200
MODE C -	23	546
	24	546
	25	280
	27	280
MODE D -	23	200
	24	280
	25	280
	27	280

3. MODE A -
   A. MAXIMUM CHARGING WATER PRESSURE SHALL BE 1600 PSIG NOMINAL. ACCUMULATOR PRECHARGE PRESSURE SHALL BE 575 PSIG NOMINAL, 580 PSIG MAXIMUM, AT 70° F.
   B. DELETED
   C. LOCATION 20 - THE CRD COOLING WATER PRESSURE SHALL NOT BE LESS THAN PR + 15 PSIG. FOR THE CONDITIONS INDICATED.
   D. LOCATION 23 - MAXIMUM DRIVE COOLING REQUIREMENTS WILL NOT EXCEED 0.34 GPM/DRIVE FOR THE CONDITIONS LISTED. MINIMUM DRIVE COOLING REQUIREMENTS WILL NOT BE LESS THAN 0.20 GPM/DRIVE.
4. MODE B -
   A. LOCATION 13 AND 14 - INSERT VALVE F007-A CLOSES ON DRIVE INSERT SIGNAL. WITHDRAW VALVE F007-B CLOSES ON DRIVE WITHDRAW SIGNAL BUT DOES NOT STAY CLOSED DURING SETTLING.
   B. LOCATION 18 - THE CRD DRIVE WATER PRESSURE SHALL NOT BE LESS THAN PR + 250 PSIG. FOR THE CONDITIONS INDICATED.
5. MODE C -
   A. DELETED
   B. THE 546 DEG. F. TEMPERATURE LISTED IN NOTE 2 FOR MODE C POSITIONS 23 AND 24 SHALL BE USED ONLY IN DETERMINING THE MINIMUM PIPE WALL THICKNESS IN VICINITY OF THE DRIVE HOUSING AND NOT IN DETERMINING STRESSES DUE TO THERMAL EXPANSION. IN DETERMINING MINIMUM WALL THICKNESS IT MAY BE ASSUMED THAT THIS TEMPERATURE OCCURS LESS THAN 1 PERCENT OF THE OPERATING LIFE OF THE SYSTEM. SEE THE CRD HYD. SYSTEM DESIGN SPECIFICATION TO DETERMINE CYCLIC STRESSES DUE TO THERMAL EXPANSION.
   C. LOCATION 21 TO 22 - THE PRESSURE DROP FROM LOCATION 21 TO 22 SHALL NOT EXCEED 435 PSI AT 90 GPM FOR ANY CRD.
   D. LOCATION 23 - A NEGATIVE FLOW RATE INDICATES FLOW FROM THE REACTOR THROUGH THE DRIVE SEAL, INTO THE CRD. THE MAXIMUM LEAK RATE FROM THE THE REACTOR CAN REACH 10 GPM PER DRIVE.
   E. LOCATION 24 TO 25 - THE PRESSURE DROP FROM LOCATION 24 TO 25 SHALL NOT EXCEED 162 PSI AT 30 GPM FOR ANY CRD.
   F. RESPONSE TIME OF FCV-F002 IS SUCH THAT SCRAM IS COMPLETED BEFORE FCV-F002 STARTS TO CLOSE.
   G. SCRAM DRAIN VALVE F011 AND VENT VALVE F010 CLOSE WITH A SCRAM SIGNAL.
6. MODE D -
   A. DELETED
   B. LOCATION 27 - THE SCRAM DISCHARGE VOLUME SHALL BE SIZED SO THAT THE RESULTING PRESSURE AFTER 100 PERCENT STROKE IS LESS THAN 65 PSIG.
7. MAXIMUM ALLOWABLE PUMP SUCTION PRESSURE SHALL BE 50 PSIG.
8. PROCESS DIAGRAM 112D1448 SHALL BE USED WITH AND FORM PART OF THIS PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.
9. DURING SCRAM, THIS FLOW WILL BE DIRECTED INTO THE SCRAM DISCHARGE VOLUME. FOLLOWING SCRAM, THIS FLOW WILL DECLINE AS VALVE F002 CLOSES AND AS THE SCRAM DISCHARGE VOLUME PRESSURIZES TO EQUAL THE REACTOR PRESSURE. AFTER THE SCRAM DISCHARGE VOLUME AND THE REACTOR VESSEL PRESSURE HAVE EQUALIZED, FLOW WILL BE DIVERTED TO THE REACTOR VESSEL VIA THE CRD WITHDRAW LINES AT A FLOW RATE DEPENDENT ON THE REACTOR PRESSURE:
   I.E. (A.) APPROX. 15 GPM AT "0" PSIG. REACTOR PRESSURE.
   (B.) APPROX. 6 GPM AT "1000" PSIG. REACTOR PRESSURE.
10. THIS VALUE APPLIES IMMEDIATELY FOLLOWING COMPLETION OF SCRAM. PRESSURE WILL SUBSEQUENTLY EQUALIZE WITH REACTOR PRESSURE.
11. DESIGN PRESSURE AND TEMPERATURE SHOWN IN "TABLE I" IS FOR INFORMATION ONLY AND IS THE BASIS FOR DESIGN OF BWRS SUPPLIED EQUIPMENT. ESTIMATED LINE SIZES ARE FOR INFORMATION ONLY. ACTUAL LINE SIZES AS DETERMINED BY THE PIPING DESIGNER SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.
12. ALL VALUES SHOWN IN MODES A, B, C, AND D ARE NOMINAL UNLESS OTHERWISE NOTED.

AMENDMENT NO. 16
June 1981



NOTES:

15. A SEPARATE SUPPRESSION POOL PENETRATION (IF USED) FOR 'C' LOOP SHALL TERMINATE UNDER WATER.

16. LINE FROM RELIEF VALVES TO SLOPE DOWNWARD CONTINUOUSLY TO SUPPRESSION POOL.

17.

18. ALLOW ADEQUATE PIPING SURFACE AREA FOR COOLING OF PUMP C003.

19. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET REF. 18.

20. EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING. THE PIPING DESIGNER SHALL SIZE ALL PIPES IN CONFORMANCE WITH THE SYSTEM DESIGN SPEC AND PROCESS DIAGRAM.

NOTES:

1. INSTRUMENT LINE DESIGN & VALVING MUST COMPLY WITH INSTRUMENT PIPING SPECIFICATION REFERENCE 9.

2. THE METHOD OF MOUNTING LOCAL INSTRUMENTS IS TO BE DETERMINED BY OTHERS.

3. PIPING HIGH POINT VENTS & LOW POINT DRAINS ARE TO BE ADDED AS NECESSARY.

4. FLUSHING CONNECTIONS SHALL BE PROVIDED IN ACCORDANCE WITH REF 2. TEMPORARY STRAINER SCREENS SHALL BE PROVIDED ON THE SUCTION SIDE OF ALL PUMPS IN ACCORDANCE WITH REF 2.

5. DISCHARGE LINES FOR COOLING WATER TO BE ROUTED UPSTREAM OF SERVICE WATER RADIATION MONITORS.

6. EQUIPMENT & INSTRUMENTS ARE PREFIXED BY SYSTEM NO. (E12) UNLESS OTHERWISE NOTED.

7. ALL MOTOR OPERATED VALVES ARE AC OPERATED UNLESS OTHERWISE NOTED.

8. REACTOR HEAD SPRAY LINE CONNECTION TO MAIN LINE SHALL BE AS CLOSE AS PRACTICIBLE TO VALVE F016.

9. RECOMMENDED. BUT NOT REQUIRED.

10. VALVE SHOULD BE INSTALLED WITH PACKING GLAND(S), UPSTREAM SIDE OF DISK.

11. VALVES F005, F025ABC, F030 & F088 SHALL BE 1" RELIEF VALVES.

12. BETWEEN VALVES MO F008 & MO F009 CONSIDERATION SHOULD BE GIVEN TO THERMAL EXPANSION OF THE CONTAINED WATER.

13. FOR ADDITIONAL CONTROL ROOM LIGHTS. SYSTEM ALARMS & REMOTE MANUAL SWITCHES SEE REF 3.

14. PROVISIONS FOR CONTAINMENT ISOLATION BY OTHERS TO BE IN ACCORDANCE WITH CURRENT LICENSING REQUIREMENTS.

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NO.

REFERENCE DOCUMENTS	MPL ITEM NO.
1. NUCLEAR BOILER SYS FCD	B22-1030
2. CLEANING OF PIPING & EQUIPMENT	A62-4140
3. RESIDUAL HEAT REMOVAL FCD	E12-1030
4. HIGH PRESSURE CORE SPRAY P&ID	E22-1010
5. RCIC SYSTEM P&ID	E51-1010
6. RADWASTE SYSTEM P&ID	G11-1010
7. PROCESS RADIATION MONITORING SYSTEM	D17-1010
8.	
9. PROCESS INSTR	A62-4070
10. RHR SYSTEM PROCESS DATA	E12-1020
11. RHR SYS DESIGN SPEC	E12-4010
12. EMERGENCY EQUIPMENT COOLING WATER	A62-4230
13. RCIC SYS FCD	E51-1030
14. REAC RECIRC SYS P&ID	B35-1010
15. NUCLEAR BOILER SYS P&ID	B22-1010
16. LOW PRESSURE CORE SPRAY P&ID	E21-1010
17. LEAK DETECTION SYSTEM IED	E31-1010
18. RHR SYS IDS	E12-3050

SUPPORTING DOCUMENTS	MPL ITEM NO.
1. PIPING & INSTRUMENT SYMBOLS	A42-1010
2. PRESSURE INTEGRITY OF PIPING & EQUIPMENT PRESSURE PARTS	A62-4030

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RHR P&ID

FIGURE 5.4-13a

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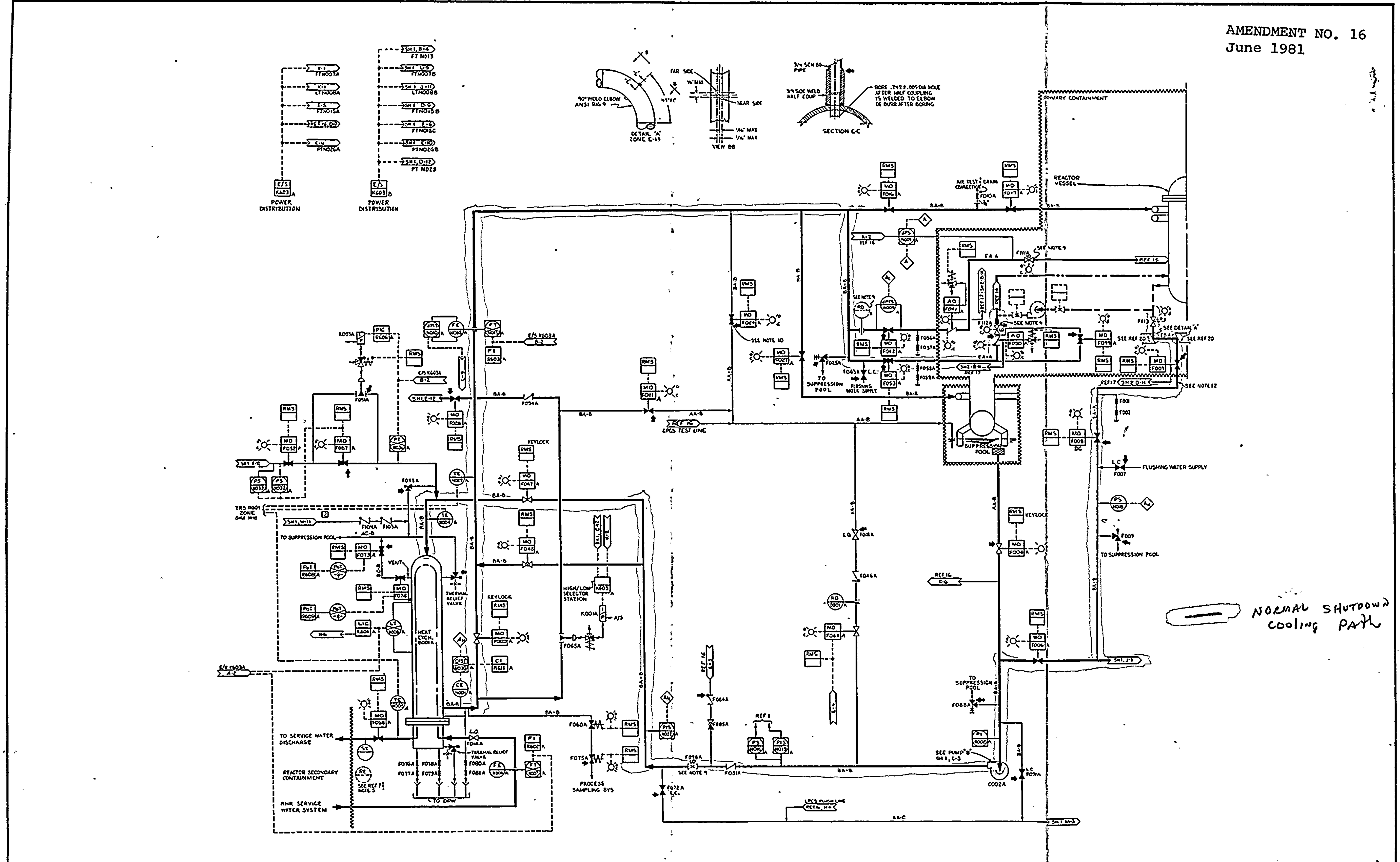

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RHR P&ID	FIGURE 5.4-13b

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FCF238X184AD (E12-1020)

MODE F (SEE NOTE 5)

POSITION ○	1	2	3	4	5	6	49	7	8	9	10	13	24	1
FLOW-GPM	—	7450	←									→	7450	—
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X			14.7
TEMP °F	—	120/40	←									→	120/40	—
MAX PRESS DROP-FEET	←					TDH=267								→

MODE A-1 (SEE NOTE 3 & 14)

POSITION ○	1	2	3	4	5	6	7	8	9	10	16	11	29
FLOW-GPM	—	7450	←								→	7450	—
PRESS-PSIA	14.7/14.7	X	X	X	X								14.7/14.7
TEMP °F	—	170/40	←								→	170/40	—
MAX PRESS DROP-FEET	←				TDH=267				ΔH=17			ΔH=40	→

MODE G (Rx PRESS 135 PSIG)

POSITION ○	29	25	26	4	5	6	43	24	1		1	2 BC	3 BC	4 BC	5 BC	6 BC	42	43 B	24 B	1
FLOW-GPM	—	550	←				→	550	—		—	550	←					→	550	—
PRESS-PSIA	150	X	X	X	X	X	X		14.7		14.7	X	X	X	X				X	14.7
TEMP °F	—	358	←				→	358	—		—	125/40	←					→	125/40	—
MAX PRESS DROP-FEET	←			TDH=650 SEE NOTE 22					→		←				TDH=650 SEE NOTE 22					→

MODE S

POSITION ○	1	2 ABC	3 ABC	4 ABC	5 ABC	6 ABC	18 AB	19 AB	20 AB	9 ABC	10 ABC	46 ABC	11 ABC	48 AB	21 AB	50 AB	27 AB	51	33	52 AB	53 AB
FLOW-GPM	N/A	←																		→	N/A
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	1015	X	155	X	1015	X	1015	X	X	X
TEMP °F	90/40	←									90/40	547	90/40	135	90/40	547	90/40	547	90/40	90/40	90/40
MAX PRESS DROP-FEET																					

MODE S (CONT'D)

POSITION ○	54	51	55 A,B		34	35	36 AB	17 AB	16 AB		20 AB	19 AB	18 AB	41	42
FLOW-GPM	N/A	N/A	N/A		N/A	←		→	N/A		N/A	←	→		N/A
PRESS-PSIA	X	X	X		X	X	X	X	X		X	X	X	X	X
TEMP °F	AMB	AMB	AMB		AMB	AMB	AMB	90/40	90/40		90/40	90/40	AMB	AMB	AMB
MAX PRESS DROP-FEET															

MODE C-3 (4 HRS, 135 PSIG R_X PRESS)

POSITION ○	29	64	65	34	35	36 AB	37 AB	17 AB	18 AB	19 AB	20 AB	38 AB	39 AB	40 AB	41	42				60 AB	61 AB	
FLOW (SEE NOTE 15)	—	89.6	114.4	←	114.4	522	←	→	572	116	←		→	116	232	232				7400	7400	
PRESS-PSIA	150					X	X	X	X	X	X	X	X	X	X	59.7/14.7				X	X	
TEMP °F	—	355.4	←					→	355.4	140	←			→	140					(95	118.8)	
MAX PRESS DROP-PSI	←						ΔP=403/135.3									→						

HEAT REMOVAL PER STEAM CONDENSING HX 62.15 X 10⁶ BTU/HR (2 HX STEAM CONDENSING)

MODE C-4 (8 HRS, 135 PSIG R_X PRESS)

POSITION ○	29	64	65	34	35	36 A	37 A	17 A	18 A	19 A	20 A	38 A	39 A	40 A	41	42				60 A	61 A	
FLOW (SEE NOTE 15)		98.7	93.5	←				→	93.5	190	←				→	190				7400	7400	
PRESS-PSIA	150	X	X	X											X	59.7/14.7				X	X	
TEMP °F	—	355.4	←					→	355.4	140	←				→	140				(95	122.5)	
MAX PRESS DROP-PSI	←						ΔP=403/135.3									→						

HEAT REMOVAL PER STEAM CONDENSING HX 101.6 X 10⁶ BTU/HR (1 HX STEAM CONDENSING)

MODE B

POSITION ○	1	2B	3B	4B	5B	6B	7B	17B	18B	19B	20B	8B	9B	10B	13B	14	15B	21B	22	23B	1		60B	61B
FLOW-GPM	—	7900	←													→	7900	7450	—	450	—		7400	7400
PRESS-PSIA	14.7/14.7				X	X	X	X	X	X	X	X	X	X	X				14.7/14.7		14.7/14.7		X	X
TEMP °F	—	212	←					→	212	179.3	179.3	179.3	179.3	179.3	179.3	179.3	179.3	179.3	—	179.3	—		(95	128.8)
MAX PRESS DROP-FEET	←				TDH=235															→				

HEAT REMOVAL PER HX 121.7 X 10⁶ BTU/HR (1 HX OPERATING) SEE NOTE 16

INDICATES FLOW PASSING THRU ABOVE POSITIONS (TYP)

SEE NOTE 7 (TYP)

MODE C-1 (R_X PRESS 1000 PSIG)

POSITION ○	29	34	35	36 AB	37 AB	17 AB	18 AB	19 AB	20 AB	38 AB	39 AB	40 AB	41	42	43 AB	24 AB	1		60 AB	61 AB
FLOW (SEE NOTE 15)	—	187.1	187.1	935	←	→	935	190	←		→	190	380	380	190	190	—		7400	7400
PRESS-PSIA	1015	X	X	X	247	X	X	X	X	X	X	X		59.7			14.7		X	X
TEMP °F	—	546	546	546	387.7	387.7	387.7	140	←					→		140	—		(95	122.4)
MAX PRESS DROP-PSI	←	ΔP=800				←		ΔP=200			ΔP=135				→					

HEAT REMOVAL PER HX 101.4 X 10⁶ BTU/HR (2 HX STEAM CONDENSING)

MODE C-2 (R_X PRESS 1000 PSIG)

POSITION ○	29	34	35	36A	37A	17A	18A	19A	20A	38A	39A	40A	41	42		60A	61A
FLOW (SEE NOTE 15)	—	136.2	←				136.2	277	←				→	277		7400	7400
PRESS-PSIA	1015	X	X	X	214.7	X	X					X	X	59.7		X	X
TEMP °F	—	546	546	546	387.7	387.7	387.7	140	←				→	140		(95	134.8)
MAX PRESS DROP-PSI	←	ΔP=800							ΔP=135					→			

DUTY PER STEAM CONDENSING HX 117.5 X 10⁶ BTU/HR (1 HX STEAM CONDENSING)

MODE C-2 (CONT'D) (SEE NOTE 14)

POSITION ○	1	2B	3B	4B	5B	6B	7B	17B	18B	19B	20B	9B	10B	13B	43B	24B	1		60B	61B
FLOW-GPM	—	7450	←												→	7450	—		7400	7400
PRESS-PSIA	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	14.7		X	X
TEMP °F	—	120/70	←				→	120/70	103.1	103.1	103.1	103.1	103.1	103.1	103.1	—			(95	103.2)
MAX PRESS DROP-FEET	←				TDH=267											→				

HEAT REMOVAL PER POOL COOLING HX 30.15 X 10⁶ BTU/HR (1 HX OPERATING)

MODE D (RX PRESS 95 PSIG)

POSITION ○	29	25	26	4 B	5 B	6 B	18 B	19 B	8 B	9 B	10 B	14	27 B	28	30	31	51	33	C9				60 B	61 B
FLOW/GPM	—	7450	7450	7450	7450	7450	7450	7450	7450	←	→	7450	7450/6580	7450/6580	870/0	870/0	870/0	870/0	—				7400	7400
PRESS-PSIA	110			X	X	X	X	X	X	X	X	X	X	X					110				X	X
TEMP °F	—	335	←			→	335	295	295	295	295	295	295	295	295	295	295	295	—				(85	125)
MAX PRESS DROP-FEET	←				TDH=267										ΔH=	17		ΔH=						

HEAT REMOVAL PER HX 148 X 10⁶ BTU/HR (1 HX OPERATING) SEE NOTE 21

SEE RCIC (REF 3)

MODE E (RX PRESS 0 PSIG)

POSITION ○	29	25	26	56	4	5	6	18 A,B	19 A,B	9	10	11	16 A,B	27 A,B	28 A,B	29		60 AB	61 AB
FLOW-GPM	—	14900/7450	14900	7450	←									→	7450	—		7400	7400
PRESS-PSIA	14.7	X	X	X	X	X	X	X	X	X	X	X				14.7		X	X
TEMP °F	—	125/AMB	←				→	125/AMB	113.8	113.8	113.8	113.8	113.8	113.8	113.8	—		(85	96.3)
MAX PRESS DROP-FEET	←					TDH=267	ΔH=23.3							ΔH=18		→		ΔH=23.1	

LOOPS A & B ONLY

DUTY PER HX 41.6 X 10⁶ BTU/HR (2 HX'S OPERATING)

THIS DRAWING SUPERSEDES DRAWING 731E966C REV.3 FOR HANFORD 2 (DIFFERENCES ARE DENOTED BY ENCIRCLEMENTS)

MODES:

- A-1 ACCIDENT WITH RECIRCULATION LINE BREAK IN EITHER SIDE AND THREE PUMP OPERATION. ONE STRAINER 50% PLUGGED.
- A-2 ACCIDENT WITH RECIRCULATION LINE BREAK IN EITHER SIDE AND THREE PUMP OPERATION. ONE STRAINER 50% PLUGGED. (10 MIN)
- B POST ACCIDENT CONTAINMENT SPRAY WITH HEAT REJECTION WITH ONE PUMP OPERATION AND STRAINER 50% PLUGGED. (PEAK SUPPRESSION POOL TEMPERATURE)
- C-1 REACTOR ISOLATION = 2HX STEAM CONDENSING (1/2 HR)
- C-2 REACTOR ISOLATION = 1HX STEAM CONDENSING AND 1HX POOL COOLING (1 1/2 HR)
- C-3 REACTOR ISOLATION = 2HX STEAM CONDENSING (4 HR)
- C-4 REACTOR ISOLATION = 1HX STEAM CONDENSING (8 HR)
- D NORMAL SHUTDOWN AFTER BLOWDOWN TO MAIN CONDENSER (SEE NOTE 11).
- E CONTINUATION OF NORMAL SHUTDOWN FROM PLANT MODE "D" (AT 0 PSIG) AND FUNCTIONAL PUMP TEST AFTER SHUTDOWN.
- F RHR SYSTEM TEST DURING PLANT OPERATION.
- G MINIMUM FLOW BYPASS MODE.
- S SYSTEM ON STANDBY DUTY

LEGEND

- R_X PRESS — REACTOR [illegible]
- TDH — TOTAL DYNAMIC [illegible]
- SHUTOFF HEAD
- ΔH — HEAD LOSS
- ΔP — PRESSURE LOSS

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. RESIDUAL HEAT REMOVAL SYSTEM P&ID	E12-1010
2. RESIDUAL HEAT REMOVAL SYSTEM DESIGN SPEC	E12-4010
3. RCIC PROCESS DIAGRAM	E51-1020
4. RECIRC SYSTEM PROCESS FLOW DIAGRAM	B35-1540
5. LOW PRESSURE CORE SPRAY PFD	E21-1020

NOTES

1. PROCESS DIAGRAM 731E966 SHALL BE USED WITH AND FORM PART OF THE PROCESS DATA. IF THERE ARE ANY CONFLICTS BETWEEN THE PROCESS DIAGRAM AND THIS PROCESS DATA, THE PROCESS DATA SHALL GOVERN.
2. PIPING BETWEEN POINTS WITH EMPTY DATA BLANKS (SEE ALSO TABLE 3) SHALL BE SIZED BY OTHERS BASED ON SPECIFIED OPERATING CONDITIONS. EMPTY DATA BLANKS CAN BE FILLED IN BASED ON ACTUAL ARRANGEMENT BY OTHERS. [X] = INDICATES THE DATA IS NOT SIGNIFICANT.
3. SHOWN AS TYPICAL FOR ONE LOOP. IF LOOPS ON SIDE [illegible] ARE NOT SYMMETRICALLY ARRANGED, VALUES FOR BOTH SIDES [illegible] BE SUBMITTED.
4. ΔP VALUES FOR EQUIPMENT WITHIN GE-BWRSD SCOPE ARE AS NOTED.
5. ELEVATIONS ARE NOT INCLUDED IN ΔP, VALUES [illegible] ELEVATIONS SHALL BE INCLUDED WHEN DETERMINING "FINAL VALUES" FOR THE EMPTY DATA BLANKS.
6. [X/Y] = INDICATES MAXIMUM (X) AND MINIMUM (Y) VALUES FOR THE MODE SPECIFIED.
7. DASHED LINES INDICATE FLOW DOES NOT PASS THRU THESE POINTS.
8. THE NPSH AVAILABLE IN MODES [illegible] AT THE CENTERLINE OF THE SUCTION NOZZLE MUST EQUAL OR EXCEED [illegible] FT.
9. SERVICE WATER CROSSTIE, [illegible] FLOW 300 GPM.
10. THIS PORTION OF PIPING TO BE SIZED [illegible] ON FLOW SHOWN ON THE FUEL POOL PROCESS DIAGRAM.
11. TABLE 3 INDICATES VALVE POSITIONS DURING VARIOUS MODES OPERATED.
12.
13.
14. TYPICAL VALUES FOR [illegible] POOL TEMP SHOWN. FINAL TEMPERATURES DEPEND ON [illegible] WATER [illegible] POOL WATER VOLUME.
15. WATER FLOWS ARE IN GPM, STEAM FLOW [illegible] 1000 [illegible].
16. HX DUTY BASED UPON 7450 GPM [illegible].
17. FOR LINE SIZING INFORMATION SEE REF. 5.
18. THE WEIGHT OF WATER IN THE SHUTDOWN [illegible] PIPING, INCLUDING THE HEAT EXCHANGERS [illegible] EXCEED 225,000 LBS AT 70°F TO PREVENT [illegible] CONTROL NEUTRON ABSORBER BELOW [illegible].
19.
20. PIPING SYSTEM DESIGN PRESSURE [illegible] LINE SIZES ARE FOR INFORMATION [illegible] PRESSURE & TEMPERATURE AND LINE SIZES AS [illegible] OTHERS SHALL MEET THE PROCESS DATA HYDRAULIC REQUIREMENTS.
21. HEAT EXCHANGER CAPABILITY SHOWN [illegible] SERVICE WATER OUTLET TEMP. OF 325°F.
22. PUMP SHUTOFF HEAD 800 FT. MAX.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RESIDUAL HEAT REMOVAL SYSTEM
PROCESS DATA SHEET 1

FIGURE 5.4-14b

SEE NOTE 20

POSITION ⬡	3.1	3.2	4	5	6	7	8	9	10	47	16	46	11	7	7J	18	19	20	8	3.2 C	4 C	5 C	6	9	10	46	11	25	54.1	26	4 A,B	5 A,B	16	50	27
DESIGN PRESS. IN PSIG	125	220		RHR PUMP	500							1250		500		RHR Hx	500			220		RHR (PUMP)	500			1250		SEE REF 4 P75/1-2	220				500	SEE REF 4 P75/1-16	
DESIGN TEMP IN °F	212	358			358						480	575		358			358			358			212			575			358				480		
ESTIMATED LINE SIZE	24"				18"						12"			18"			18"			24"			18"			12"		20"		18"	24"		12"		
	LPCI LINE (VIA RHR Hx BYPASS)													Hx LINE						LPCI LINE (DIRECT TO REACTOR)								SHUTDOWN SUCTION					SHUTDOWN RETURN LINE		

POSITION ⬡	34	35	36	37	17	18	19	20	38	39	40	40.1	41	42	14	30	31	51	66	40	40.2	44	47	48	21	15	52	23	15	53	44	45	24
DESIGN PRESS. IN PSIG	1250				500		500			500		125			500			SEE (REF. 3)		500	125		500			500			500	125			
DESIGN TEMP IN °F	575				480		480			140		140			358					140	212		358	212		358	212		358	212			
ESTIMATED LINE SIZE	10"		8		18		18"	4"							6"					4"			16"			4"			18"				
	STEAM TO HX						CONDENSATE FROM HX								HEAD SPRAY LINE					CONDENSATE TO POOL			DRYWELL SPRAY			SUPPRESSION POOL SPRAY			SYSTEM TEST LINES (LOOPS A & B)				

POSITION ⬡	56	56.1	4	49	49J	45	6	61	62	45	6	61	45
DESIGN PRESS. IN PSIG	220			500	125		500	125			500	125	
DESIGN TEMP IN °F	358			212			212				358	212	
ESTIMATED LINE SIZE	18"			18"			4"		4"		4"		
	SYSTEM TEST LINES (LOOP C)						MIN FLOW BYPASS LINES (LOOP C)				MIN FLOW BYPASS (LOOPS A,B)		

TABLE 1 - VALVE POSITION CHART

SEE NOTE 11

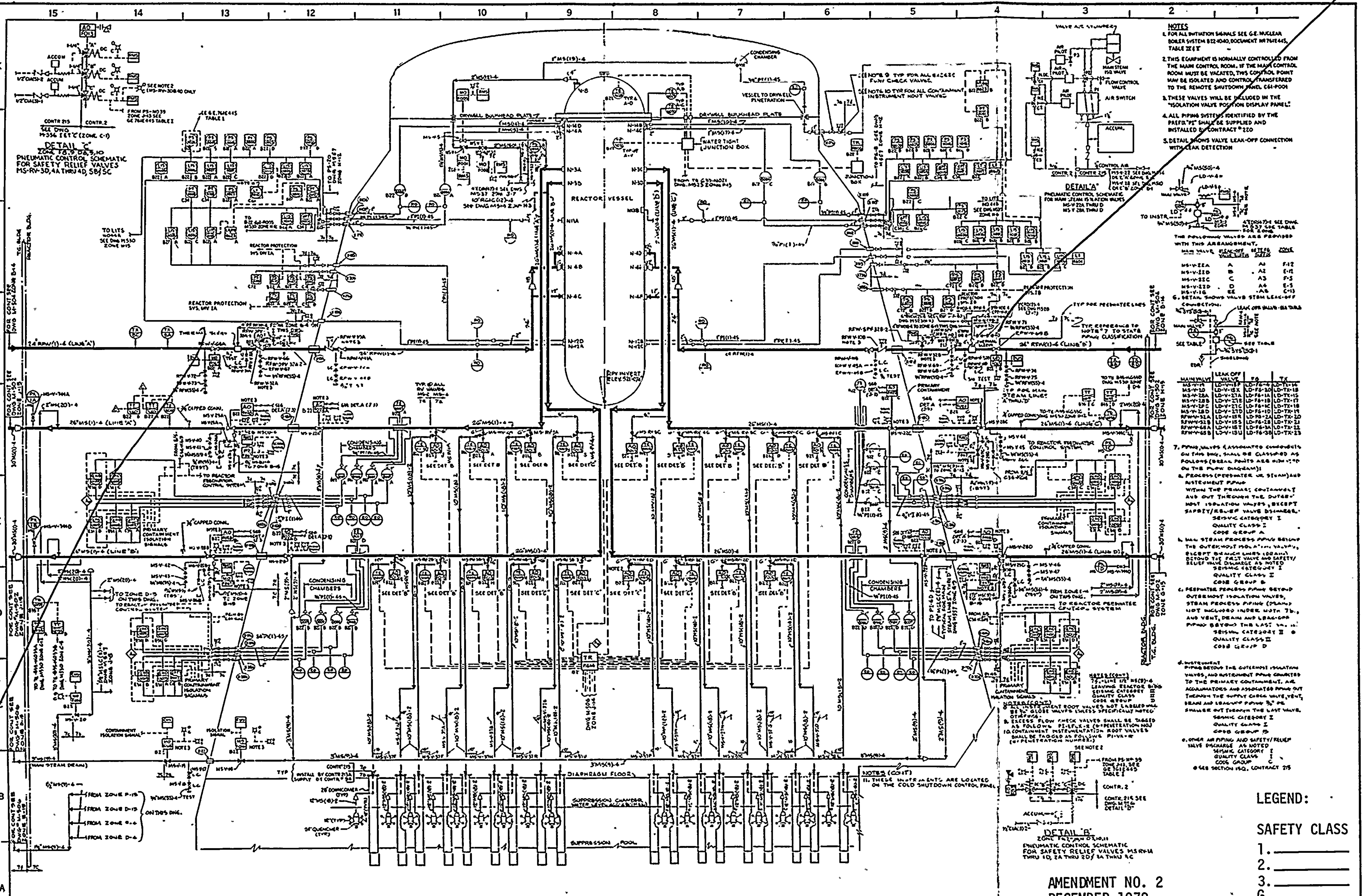
F004 A & B / F004 C	[illegible]	STRAINER AB	STRAINER C
MODE A-1	0 0 0 0 0	1-50% P	
MODE A-2	0 0 0 0 0	1-50% P	
MODE B	0 0 0 0 0 0 0	50%	---
MODE C-1	0 T T 0 0 0	—	—
MODE C-1	0 T T 0 0 0	—	—
MODE C-2	0 T T 0 0 0	—	—
C-2 CONTD	0 0 0 T-C 0	CLEAR	—
MODE C-3	0 0 0 T 0 0 0	—	—
MODE C-4	0 0 0 T 0 0 0	—	—
MODE D	0 0 T-C 0 T 0 T 0 0 0 0		
MODE E	0 0 0 0-T 0 0 0 0 0 0		
MODE F	0 0 0 T-C T-C	CLEAR	CLEAR
MODE G	[illegible]	CLEAR	CLEAR
MODE S	0 0 0 0 0 0	CLEAR	CLEAR

O - VALVE OPEN
T - VALVE THROTTLED
P - STRAINER PLUGGED
T-C - VALVE THROTTLED OR CLOSED
O-T - VALVE OPEN OR THROTTLED
BLANK SPACE INDICATES VALVE IS CLOSED

TABLE 3 LIMITING LINE LOSSES

MODE A-1	6-11 PUMP DISCHARGE LINE TO RPV FLOODING PENETRATION
MODE A-2	NONE
MODE B	1-2-5 (SUCTION LINE SUPPRESSION POOL TO PUMP), 47-21 & 15-23 (CONTAINMENT SPRAY LINES PUMP DISCHARGE TO TO SPRAY HEADERS)
MODE C-1	40-44 (Hx CONDENSATE TO SUPPRESSION POOL), 41-42 (Hx CONDENSATE LINE TO RCIC LINE)
MODE C-2	20-40-41 (Hx CONDENSATE TO RCIC LINE POINT 41)
C-2 CONTD	NONE
MODE C-3	29-34-35 (STEAM LINE FROM RCIC LINE TO Hx HEADER)
MODE C-4	35-17 (STEAM LINE HEADER TO Hx)
MODE D	14-51-33-29 (VESSEL HEADER SPRAY LINE) 29-25-5 (SHUTDOWN SUCTION LINE RPV TO PUMP) 56-4 (SHUTDOWN SUCTION LINE TEST BRANCH TO PUMP)
MODE E	16-27 (SHUTDOWN DISCHARGE LINE LPCI BRANCH TO RECIRC LOOP)
MODE F	13-43-24 II 49-45 (TEST LINE TO SUPPRESSION POOL)
MODE G	6-43 (PUMP MINIMUM BYPASS LINE)

Superseded per Amdt. 22 to FSAR dtd 3-19-82
50-397

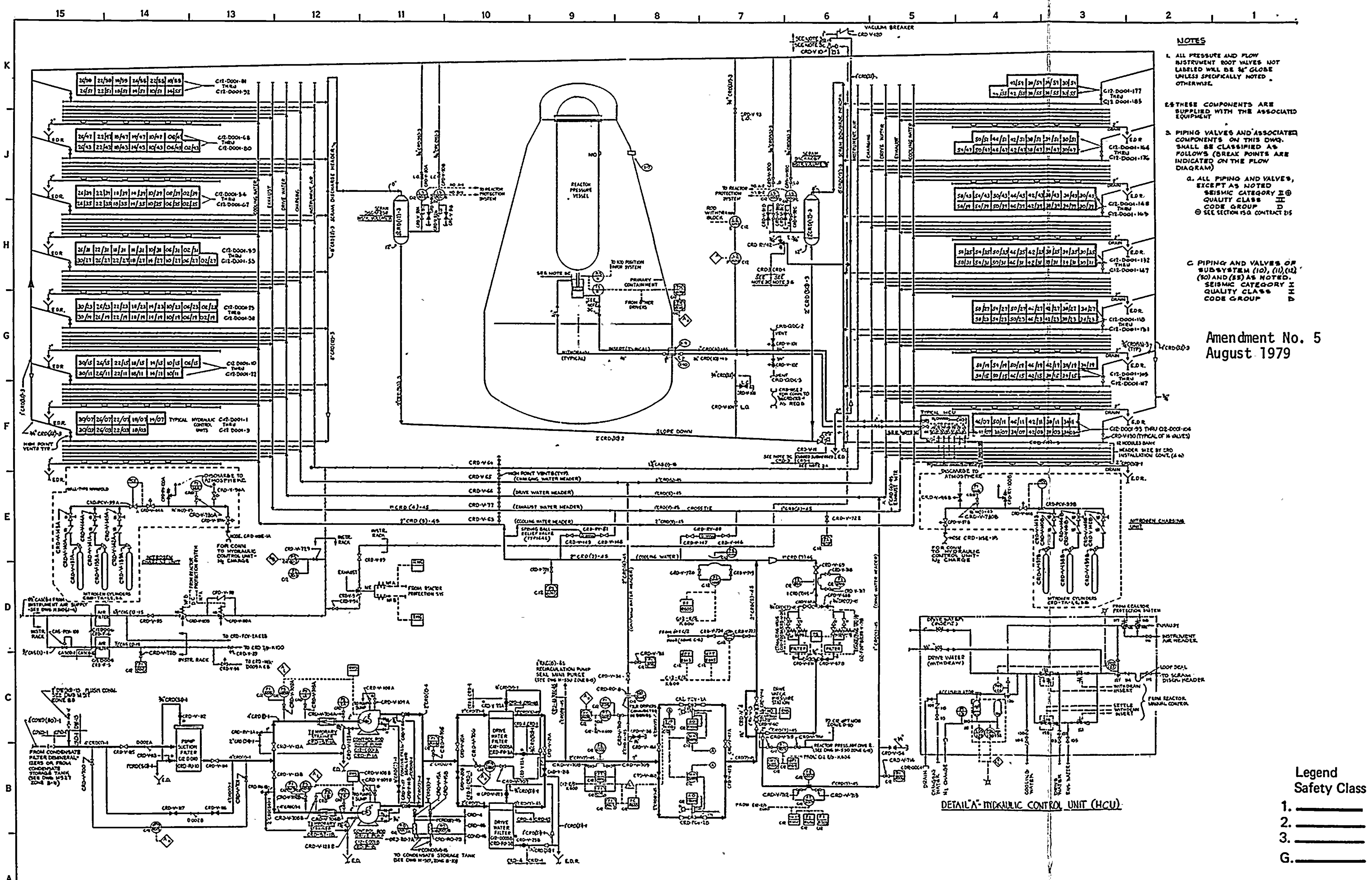


AMENDMENT NO. 2
DECEMBER 1978

LEGEND:

SAFETY CLASS
1. ______
2. ______
3. ______
G. ______

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NUCLEAR BOILER SYSTEM	FIGURE 3.2-2

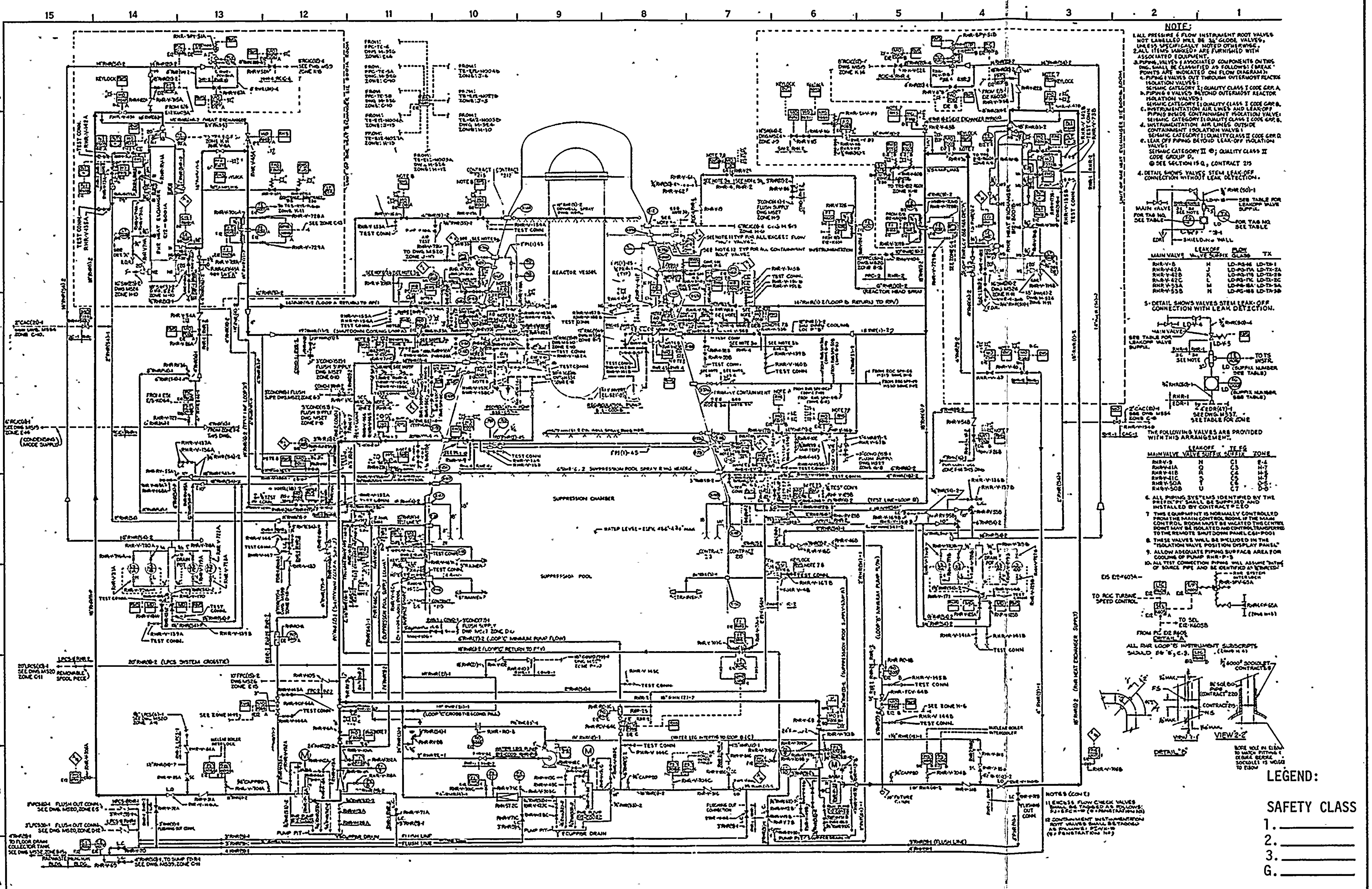


Amendment No. 5
August 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

FLOW DIAGRAM
CONTROL ROD DRIVE SYSTEM

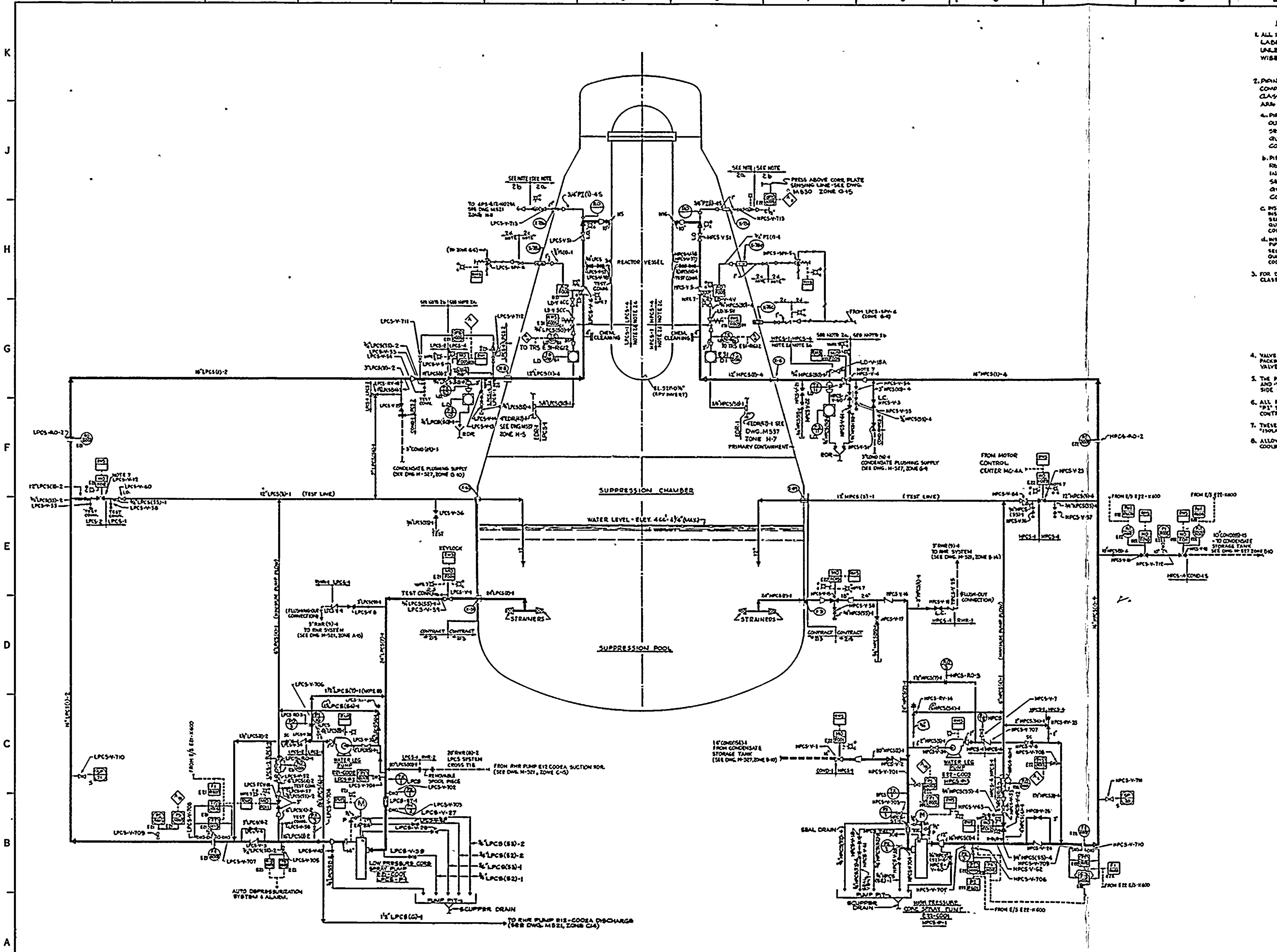
FIGURE 3.2-4



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

RESIDUAL HEAT REMOVAL SYSTEM

FIGURE 3.2-6



NOTES:

1. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.

2. PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DWG. SHALL BE CLASSIFIED AS FOLLOWS (BREAK POINTS ARE INDICATED ON FLOW DIAGRAM)

   a. PIPING AND VALVES OUT THROUGH OUTERMOST REACTOR ISOLATION VALVES
      SEISMIC CATEGORY I
      QUALITY CLASS - I
      CODE GROUP - A

   b. PIPING AND VALVES BEYOND OUTERMOST REACTOR ISOLATION VALVES PLUS ALL INSTRUMENT LINES.
      SEISMIC CATEGORY I
      QUALITY CLASS - I
      CODE GROUP - B

   c. INSTRUMENTATION AIR LINES AND LEAK-OFF PIPING INSIDE CONTAINMENT ISOLATION VALVE:
      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B

   d. INSTRUMENTATION AIR LINES AND LEAK-OFF PIPING OUTSIDE CONTAINMENT ISOLATION VALVE:
      SEISMIC CATEGORY I
      QUALITY CLASS II
      CODE GROUP D

3. FOR DRAIN OR LEAK-OFF LINES, THE PIPING CLASSIFICATION IS AS SHOWN ON DETAIL BELOW

   LAST VALVE

   E.D.

   * SAME AS FOR MAIN LINE OR VALVE, OR AS NOTED.

4. VALVE HPCS-V-23 SHALL BE INSTALLED WITH THE PACKING GLAND ON THE UPSTREAM SIDE OF THE VALVE DISC.

5. THE PACKING GLAND OF VALVE LPCS-FCV-11 AND -V-12 SHALL BE LOCATED ON THE UPSTREAM SIDE OF VALVE DISC.

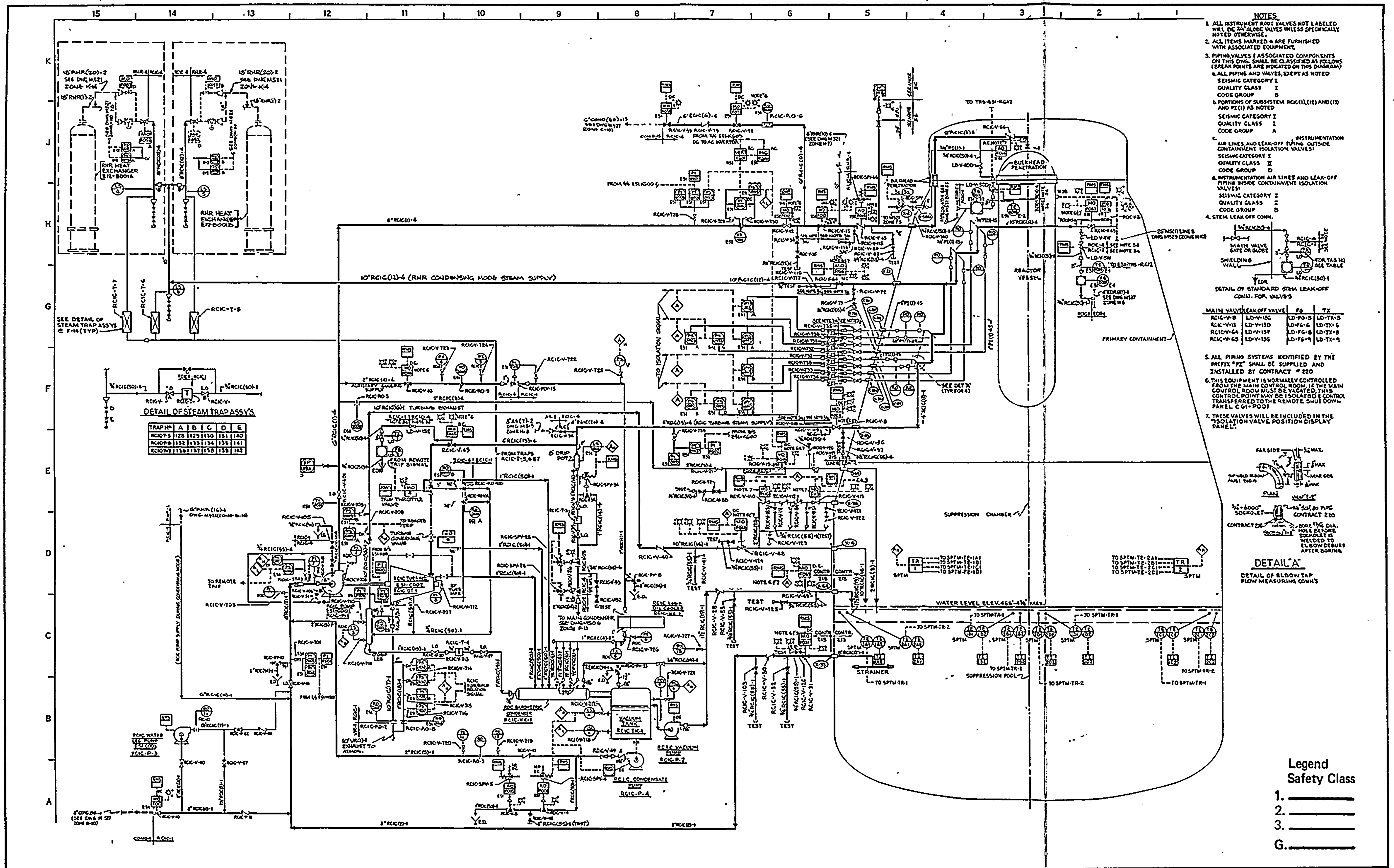
6. ALL PIPING SYSTEMS IDENTIFIED BY THE PREFIX "PI" SHALL BE SUPPLIED AND INSTALLED BY CONTRACT # 220

7. THESE VALVES WILL BE INCLUDED IN THE "ISOLATION VALVE POSITION DISPLAY PANEL"

8. ALLOW ADEQUATE PIPING SURFACE AREA FOR COOLING OF PUMP LPCS-P-2.

Legend
Safety Class
1.
2.
3.
G.

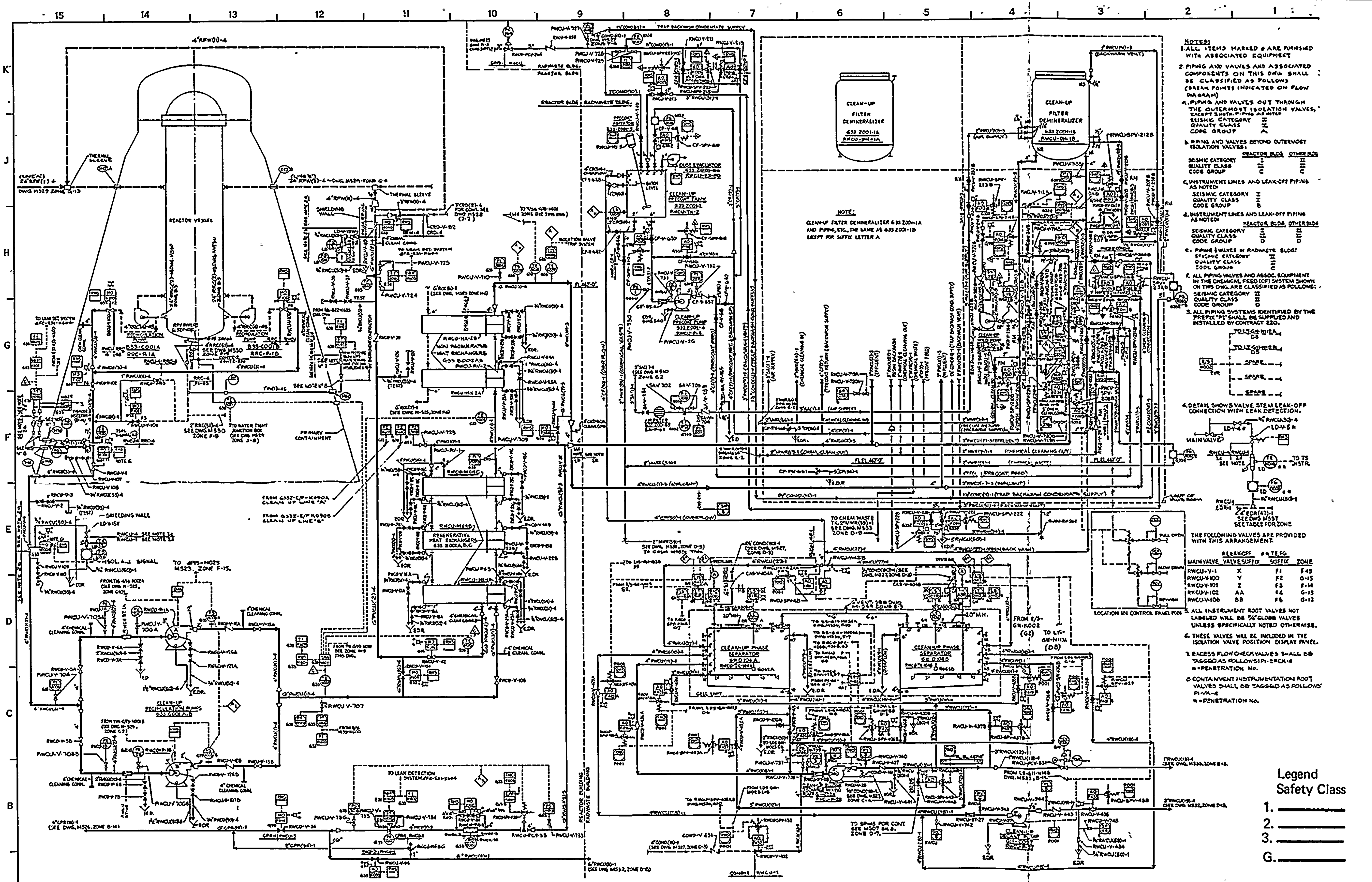
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HIGH PRESSURE CORE SPRAY AND LOW PRESSURE CORE SPRAY SYSTEMS	FIGURE 3.2-7



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR CORE ISOLATION COOLING SYSTEM

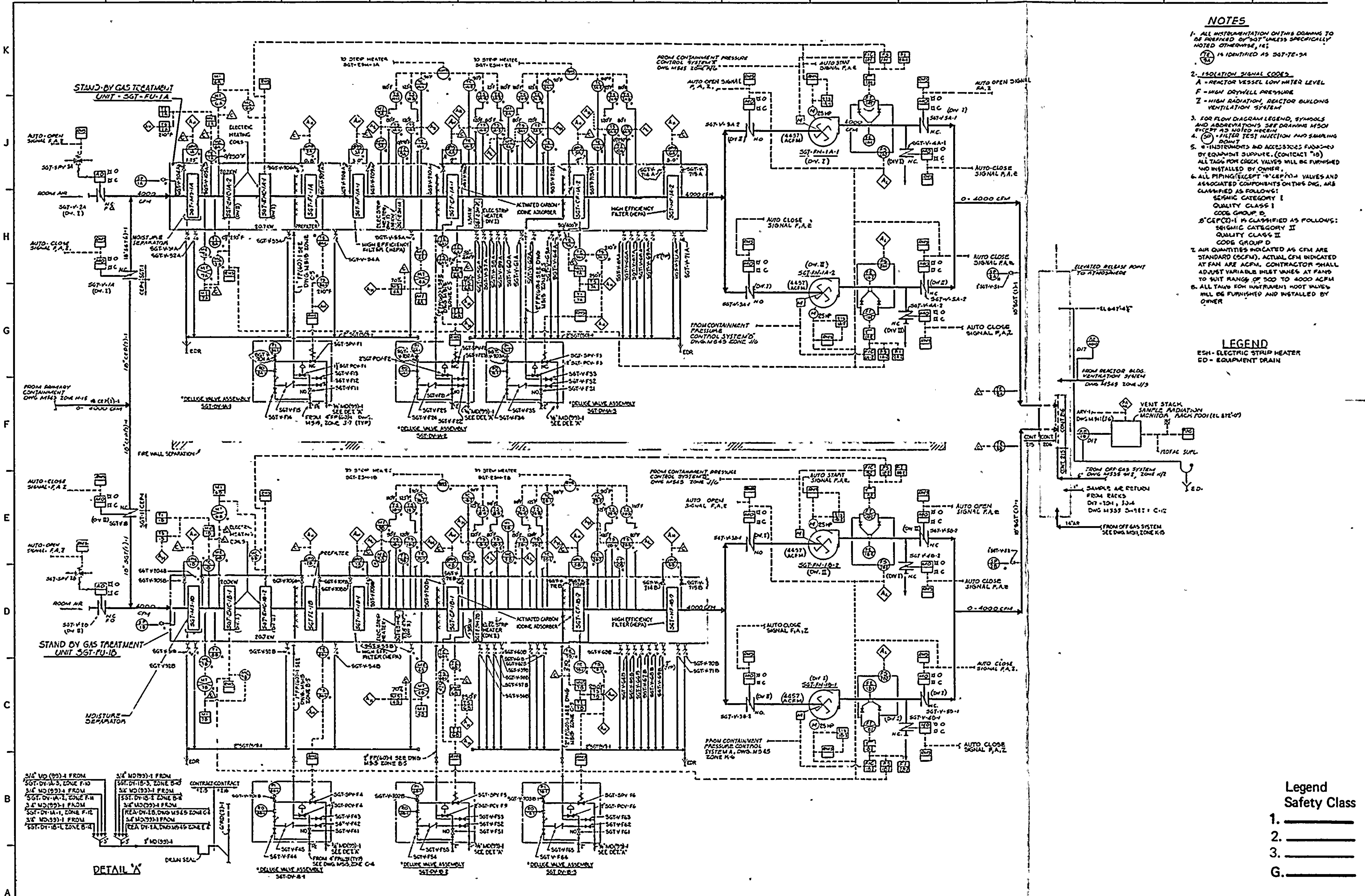
FIGURE 3.2-8



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

REACTOR WATER CLEAN-UP SYSTEM

FIGURE 3.2-11



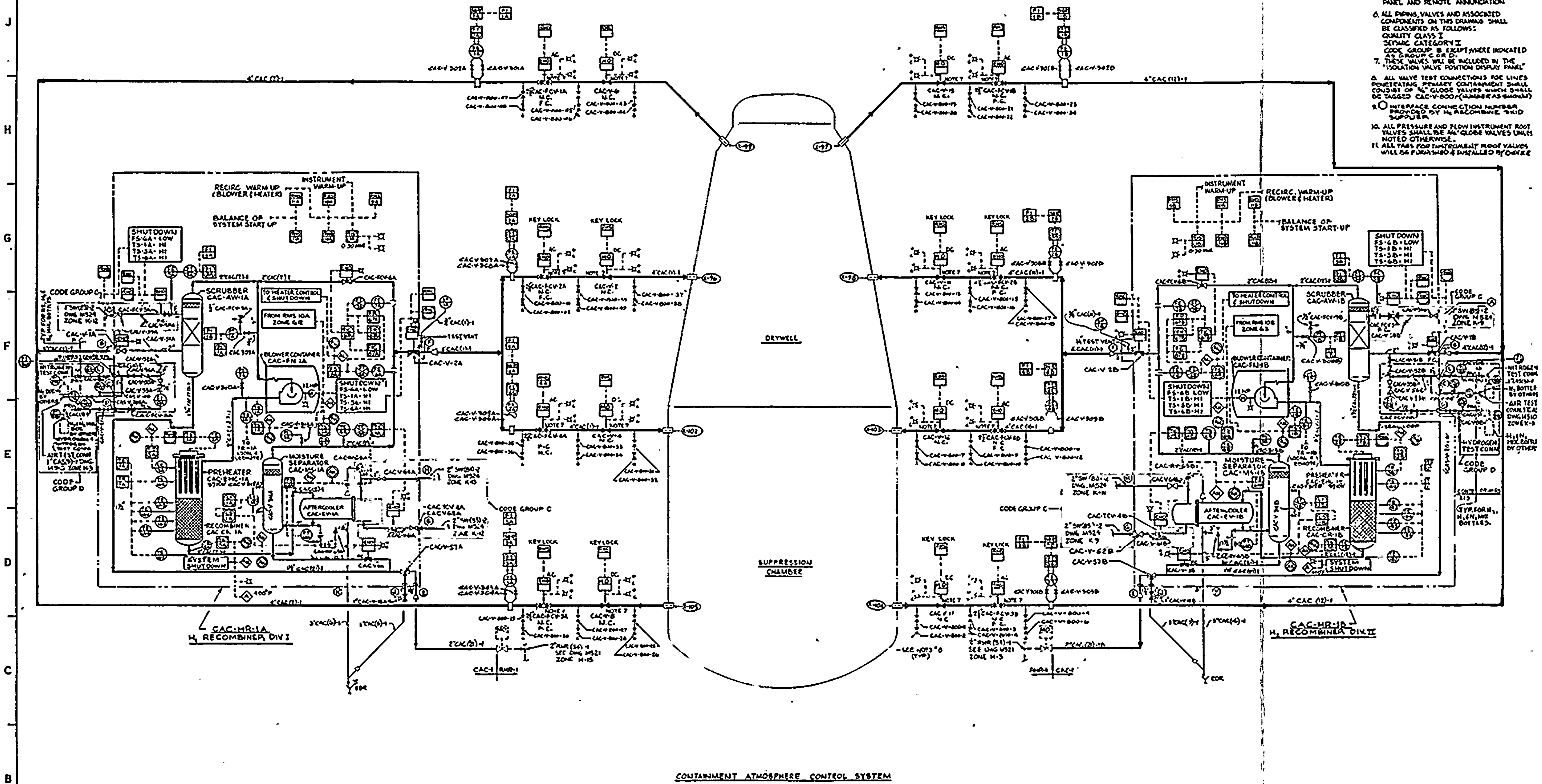
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

STANDBY GAS TREATMENT SYSTEM

FIGURE 3.2-16

NOTES:

1. ALL INSTRUMENTATION ON THIS DRAWING TO BE PREFIXED BY CAC UNLESS SPECIFICALLY NOTED OTHERWISE I.E. (PI 1) IS IDENTIFIED AS CAC-PI-1
2. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DRAWING M501
3. MOTOR OPERATED VALVES (A-C ONLY) ARE WITH SLOW CLOSURE TIME (WITH POSITION INDICATOR IN CONTROL ROOM)
4. THE UNITS ARE STARTED FROM MAIN CONTROL ROOM. FOR HYDROGEN AND OXYGEN SAMPLING AND RECORDING SYSTEMS SEE DWG. M543.
5. ALARMS INDICATE LIGHTS ON LOCAL PANEL AND REMOTE ANNUNCIATION
6. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS I
   SEISMIC CATEGORY I
   CODE GROUP B EXCEPT WHERE INDICATED AS GROUP C OR D.
7. THESE VALVES WILL BE INCLUDED IN THE "ISOLATION VALVE POSITION DISPLAY PANEL"
8. ALL VALVE TEST CONNECTIONS FOR LINES PENETRATING PRIMARY CONTAINMENT SHALL CONSIST OF ¾" GLOBE VALVES WHICH SHALL BE TAGGED CAC-V-800-(NUMBER AS SHOWN)
9. ◯ INTERFACE CONNECTION NUMBER PROVIDED BY H_2 RECOMBINER SKID SUPPLIER
10. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE ¾" GLOBE VALVES UNLESS NOTED OTHERWISE.
11. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED & INSTALLED BY OWNER

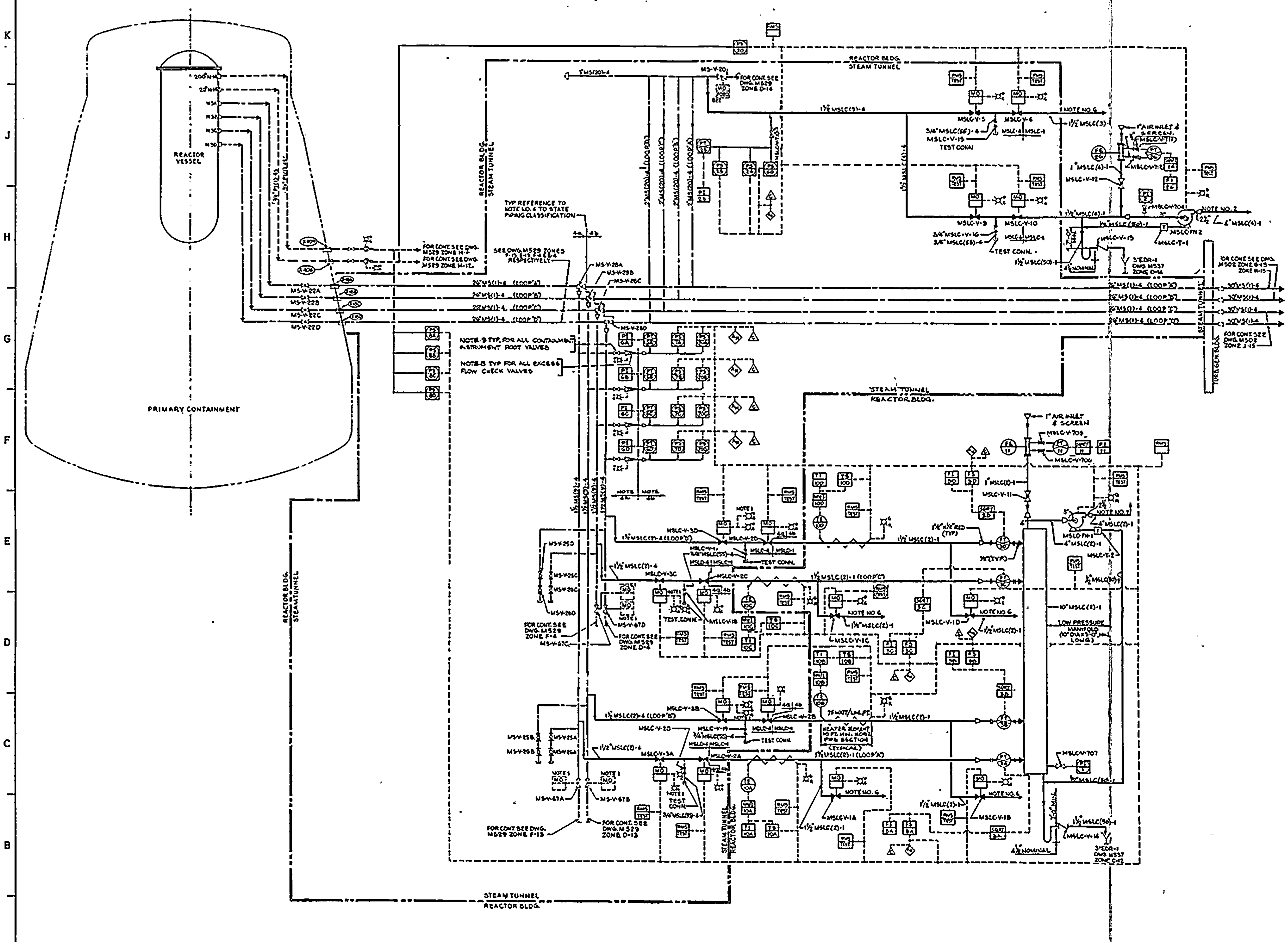


LEGEND:

SAFETY CLASS

1. ______
2. ______
3. ______
G. ______

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRIMARY CONTAINMENT ATMOSPHERIC CONTROL SYSTEM	FIGURE 3.2-17

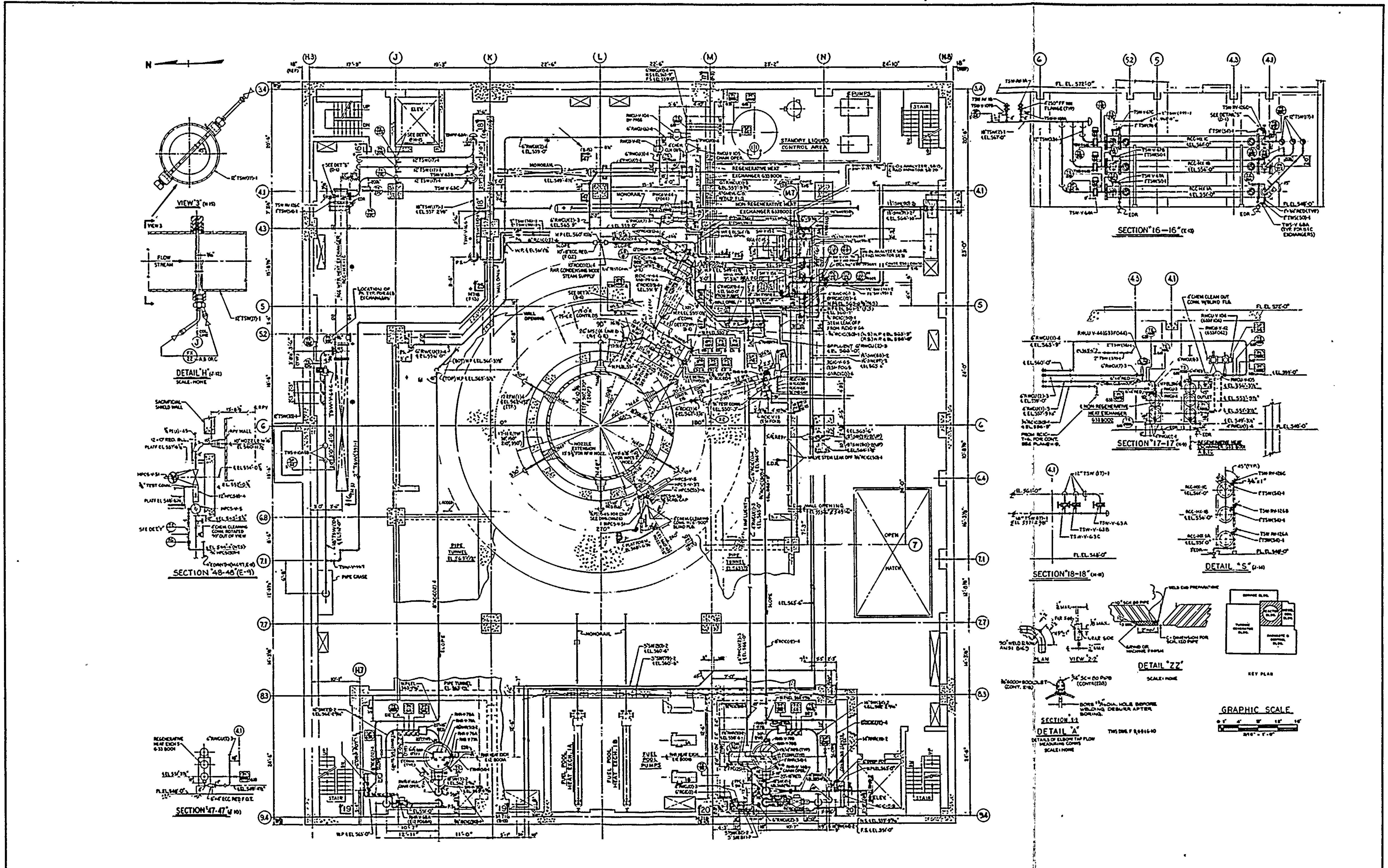


NOTE

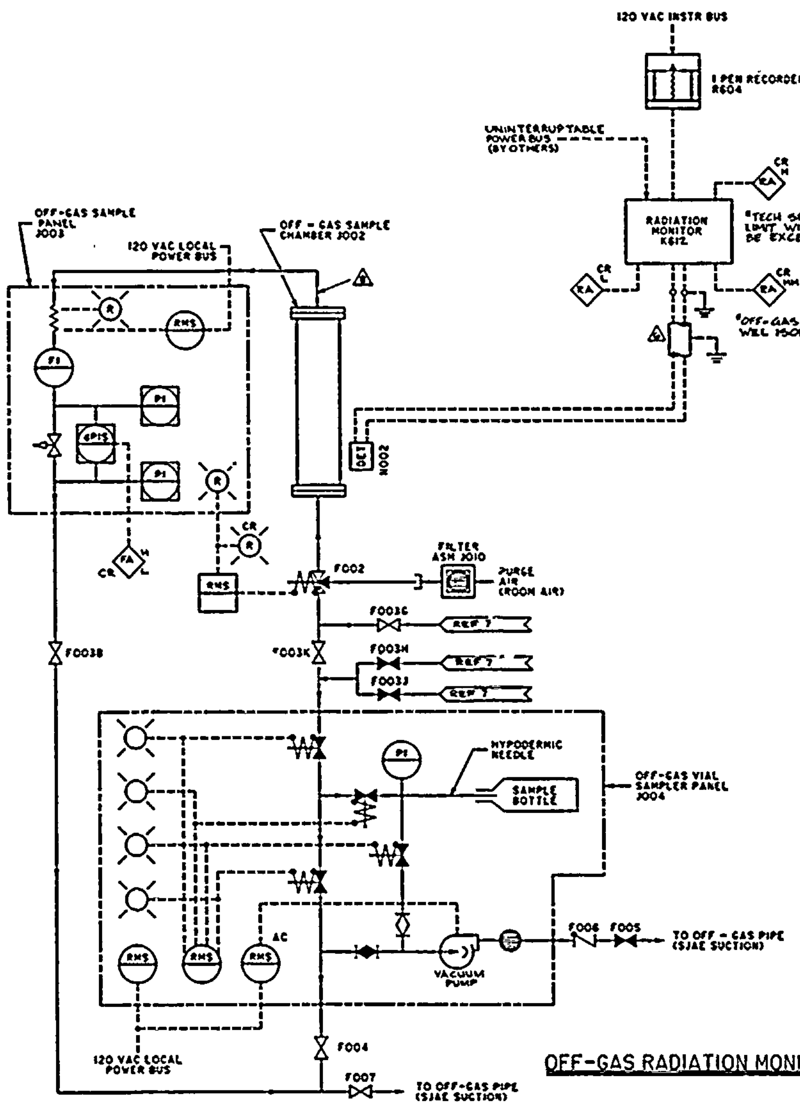
1. THESE VALVES ARE INCLUDED IN THE ISOLATION VALVE POSITION DISPLAY PANEL.
2. EXHAUST TO WITHIN 5'-0" TO 10'-0" OF STANDBY GAS TREATMENT SYSTEM INLET.
3. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE UNLESS SPECIFICALLY NOTED OTHERWISE.
4. PIPING, VALVES & ASSOCIATED COMPONETS ON THIS DWG SHALL BE CLASSIFIED AS FOLLOWS (BREAK POINTS ARE INDICATED ON THE FLOW DIAGRAM)
   a) PROCESS AND INSTRUMENT PIPING 3/4" AND LARGER WITHIN THE PRIMARY CONTAINMENT AND OUT THROUGH THE OUTERMOST ISOLATION VALVES
      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP A
   b) PROCESS & INST. PIPING BEYOND THE OUTERMOST ISOLATION VALVES
      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B
5. SLOPE ALL PIPING 1/8" PER FT TOWARDS DRAIN SEALS.
6. EXHAUST TO SAFE AREA IN REACTOR BUILDING TO AVOID HAZARD TO PERSONNEL.
7. ALL INSTRUMENTATION ON THIS DIAG. SHALL BE PREFIXED BY, MSLC.
8. EXCESS FLOW CHECK VALVES SHALL BE TAGGED AS FOLLOWS: PI-EFCX-* (* = PENETRATION No)
9. CONTAINMENT INSTRUMENTATION ROOT VALVES SHALL BE TAGGED AS FOLLOWS PI-VX-* (* = PENETRATION No)

Legend
Safety Class
1.
2.
3.
G.

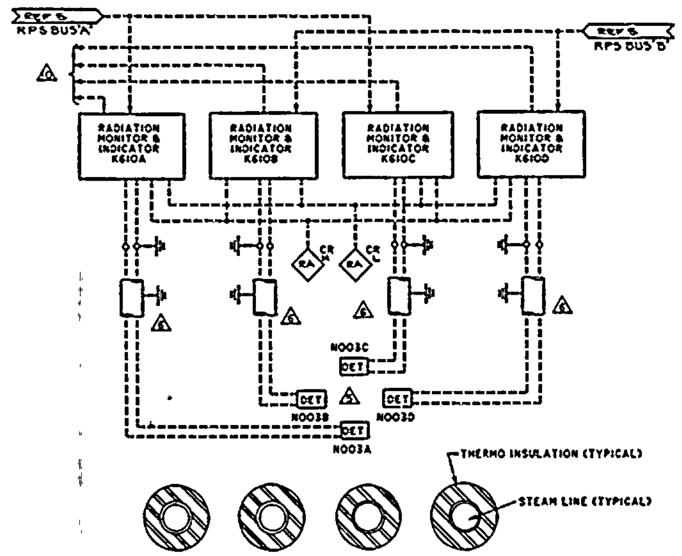
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM	FIGURE 3.2-25



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RFW, RCIC, HPCS, SW, RWCU, RRC & TSW EL. 548' REACTOR BLDG. (M715)	FIGURE 3.5-13



OFF-GAS RADIATION MONITORS & SAMPLERS



MAIN STEAM LINE RADIATION MONITORS

NOTES:

1. THE OFF GAS VENT PIPE GAS SAMPLE LINE SHALL BE 1" X 0.058" WALL THICKNESS SEAMLESS STAINLESS STEEL TUBING. THE TUBING MIN BEND RADIUS SHALL BE 20". THE TUBING LENGTH SHALL BE JOINED WITH SWAGELOK TYPE 1610-6-316 UNIONS. THE TUBING SHALL SLOPE SO THAT THE CONDENSATE WILL RUN TO DRAIN TEE.
2. A REMOVABLE SECTION SHALL BE PROVIDED NEAR THE ISOKINETIC PROBE FOR THE INSERTION OF A CHARCOAL FILTER HOLDER. THE FITTINGS ETC. SHALL PROVIDE SMOOTH TRANSITIONS WITHOUT DISCONTINUITIES OR REDUCING THE CROSS SECTIONAL AREA OF THE FLOW STREAM.
3. TEE SHALL BE UNION TEE SWAGELOK TYPE 1610-3-316.
4. ALARMS ARE ACTUATED BY RELAYS IN TRIP AUX. UNIT. DOWNSCALE ALARMS FOR LIQUID RADIATION MONITORS ARE ANNUNCIATED ON A SINGLE COMMON ANNUNCIATOR.
5. THE MAIN STEAM LINE RADIATION MONITOR DETECTORS (N003) SHALL BE LOCATED WITHIN THE STEAM LINE TUNNEL AS CLOSE AS PRACTICAL TO THE PRIMARY CONTAINMENT. THE DETECTORS SHALL BE ARRANGED SUCH THAT EACH DETECTOR WILL VIEW ALL STEAM LINES WITH APPROXIMATELY THE SAME RESPONSE. IT IS RECOMMENDED THAT THE DETECTOR OR DETECTOR ASSEMBLY BE FASTENED TO A ROD OR A PIPE AND INSERTED INTO SEALED PIPE WALLS FROM OUTSIDE THE STEAM TUNNEL. CAREFULLY ROUTE CABLES TO MINIMIZE HEAT EXPOSURE. NO LEAD SHIELDING IS REQUIRED.
6. ALL CABLES SHALL COMPLY WITH GE ENGR. SPEC REF 3.
7. ADDITIONAL ALARM IN RADWASTE BLDG (RAH) RADWASTE MONITOR ONLY.
8. DRAIN AT THE LOWER POINT OF OFF GAS SAMPLE LINE. SAMPLE LINE SHALL HAVE 2 MINUTE TRANSIT TIME TO ALLOW N^{16} DECAY.
9. TWO OUT OF TWO HIGH HIGH RADIATION (RAHH) OR INOPERATIVE TRIP (CHANNELS A AND B) SHALL:
   -- SHUTDOWN AND ISOLATE (OUTBOARD VALVE) REACTOR BUILDING VENTILATION SYSTEM.
   -- INITIATE STANDBY GAS TREATMENT SYSTEM TRAIN B.
   -- CLOSE OUTBOARD PRIMARY CONTAINMENT PURGE AND VENT VALVES.
   TWO OUT OF TWO HIGH-HIGH RADIATION (RAHH) OR INOPERATIVE TRIP (CHANNELS C AND D) SHALL:
   -- SHUTDOWN AND ISOLATE (INBOARD VALVE) REACTOR BUILDING VENTILATION SYSTEM.
   -- INITIATE STANDBY GAS TREATMENT SYSTEM TRAIN A
   -- CLOSE INBOARD PRIMARY CONTAINMENT PURGE AND VENT VALVES.
   -- ANY ONE HIGH-HIGH RADIATION TRIP (RAHH) SHALL ALARM. SEE REF 7
10. ONE HIGH HIGH RADIATION (RAHH OR INOPERATIVE TRIP OUT OF TWO ON TRIP SYSTEM "A" AND ONE HIGH-HIGH RADIATION (RAHH) OR INOPERATIVE TRIP OUT OF TWO ON TRIP SYSTEM "B" SHALL:
    -- CLOSE MAIN STEAM LINE ISOLATION VALVES.
    -- SCRAM REACTOR
    -- TURN OFF MECHANICAL VACUUM PUMP & CLOSE MECHANICAL VACUUM PUMP LINE VALVE.
    -- ANY ONE HIGH-HIGH RADIATION TRIP (RAHH) SHALL ALARM.
11. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET (SEE REF 9)
12. ANYONE UPSCALE TRIP (RAH) SHALL CLOSE BYPASS LINE VALVE, OPEN TREATMENT LINE VALVE AND ALARM.
13. ISOLATE OFF-GAS SYSTEM OUTLET AND DRAIN VALVES AND ALARM (REF 8).
14. *SUPPLIED AND MOUNTED BY OTHERS.*

SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NUMBERS

REFERENCE DOCUMENTS:	MPL ITEM NO.
1. PIPING & INST. SYMBOLS	A42-1010
2. PROCESS RADIATION MONITORING DES. SPEC	D17-4010
3. SPECIAL WIRE & CABLE	A62-4010
4. RADWASTE SYSTEM P&ID	G11THRU G16-1010
5. REACTOR PROTECTION SYS IED	C71-C72-1010
6. NUCLEAR BOILER SYS FCD	B21-B22-1030
7. OFF-GAS SYSTEM P&ID	N62-N64-1010
8. OFF-GAS SYSTEM FCD	N62-1030
9. INST. DATA SH.	D17-3050

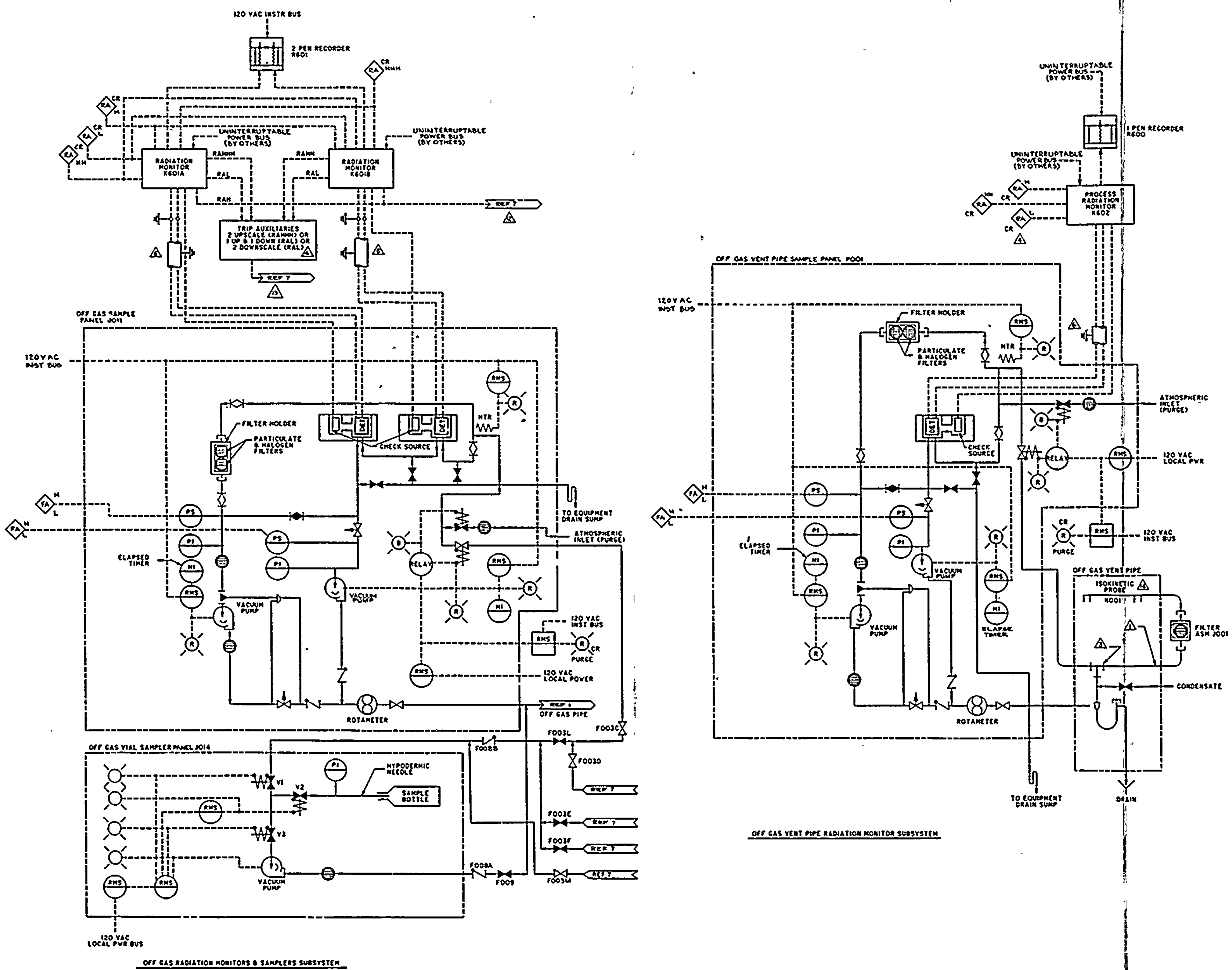
LEGEND:

SJAE	STEAM JET AIR EJECTOR
DET	DETECTOR
RAHHH	RADIATION ALARM HIGH HIGH HIGH
RAHH	RADIATION HIGH HIGH
RAH	RADIATION HIGH
RAL	DOWNSCALE OR INSTRUMENT TROUBLE
FAH/L	FLOW ALARM HIGH/LOW

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 1	FIGURE 7.2-8a

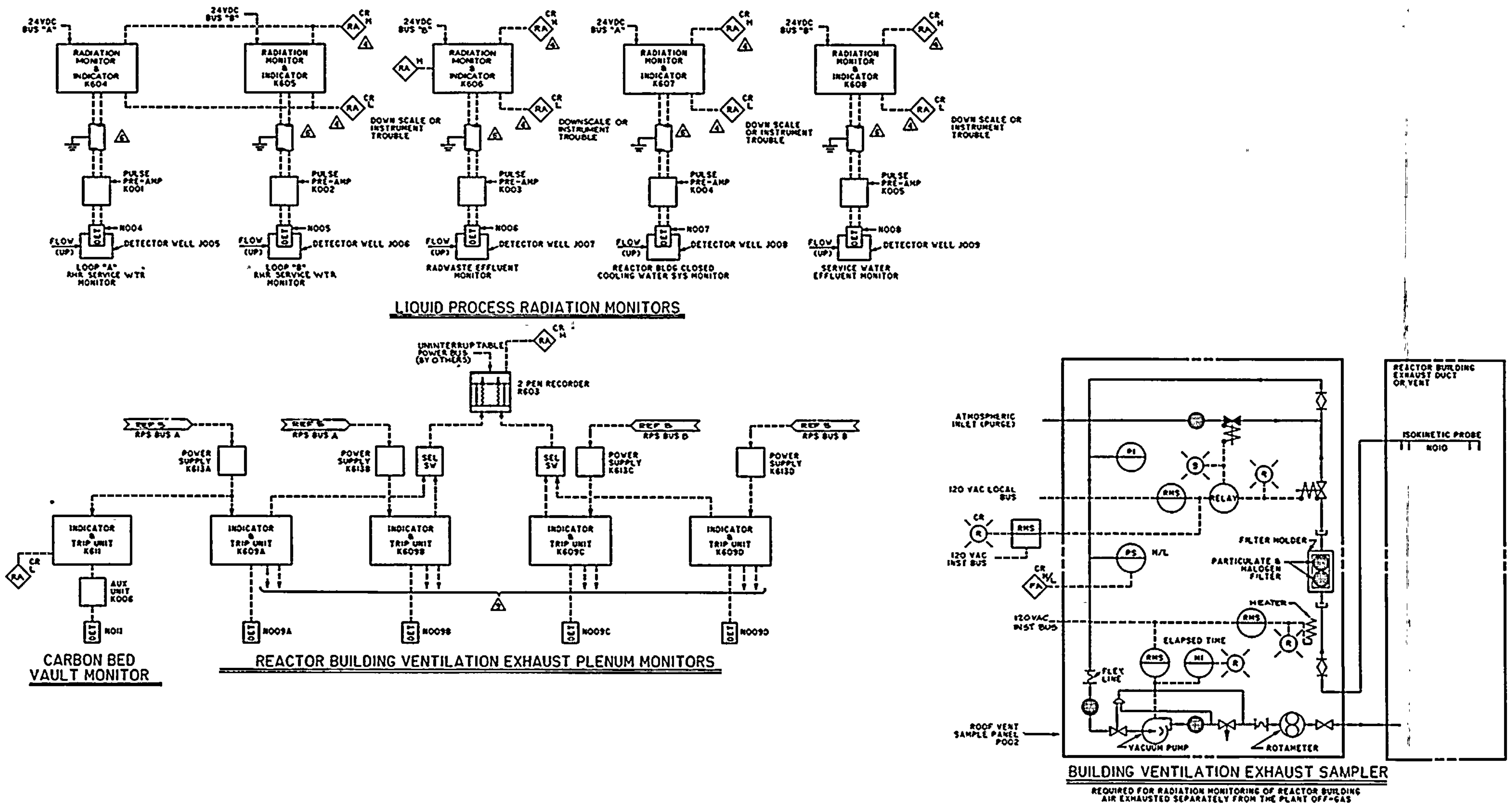
(NOT USED)

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.6-5



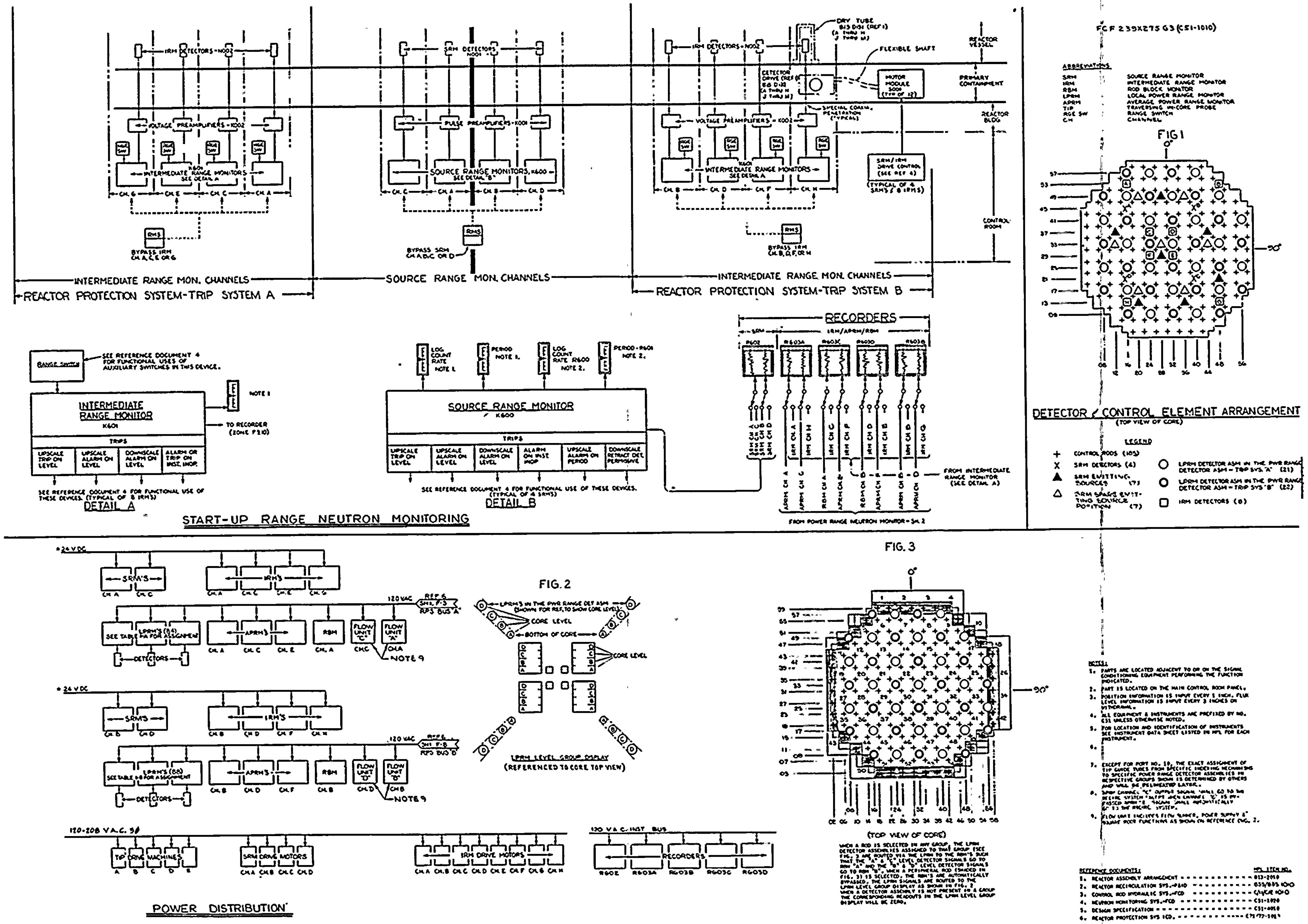
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 2	FIGURE 7.2-8b

AMENDMENT NO. 16
June 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PROCESS RADIATION MONITORING SYSTEM (RECOMB/CHARCOAL BED) IED SHEET 3	FIGURE 7.2-8c

AMENDMENT NO. 16
June 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM IED - SHEET 1	FIGURE 7.6-4a

AMENDMENT NO. 16
June 1981

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 7.3-. 9a

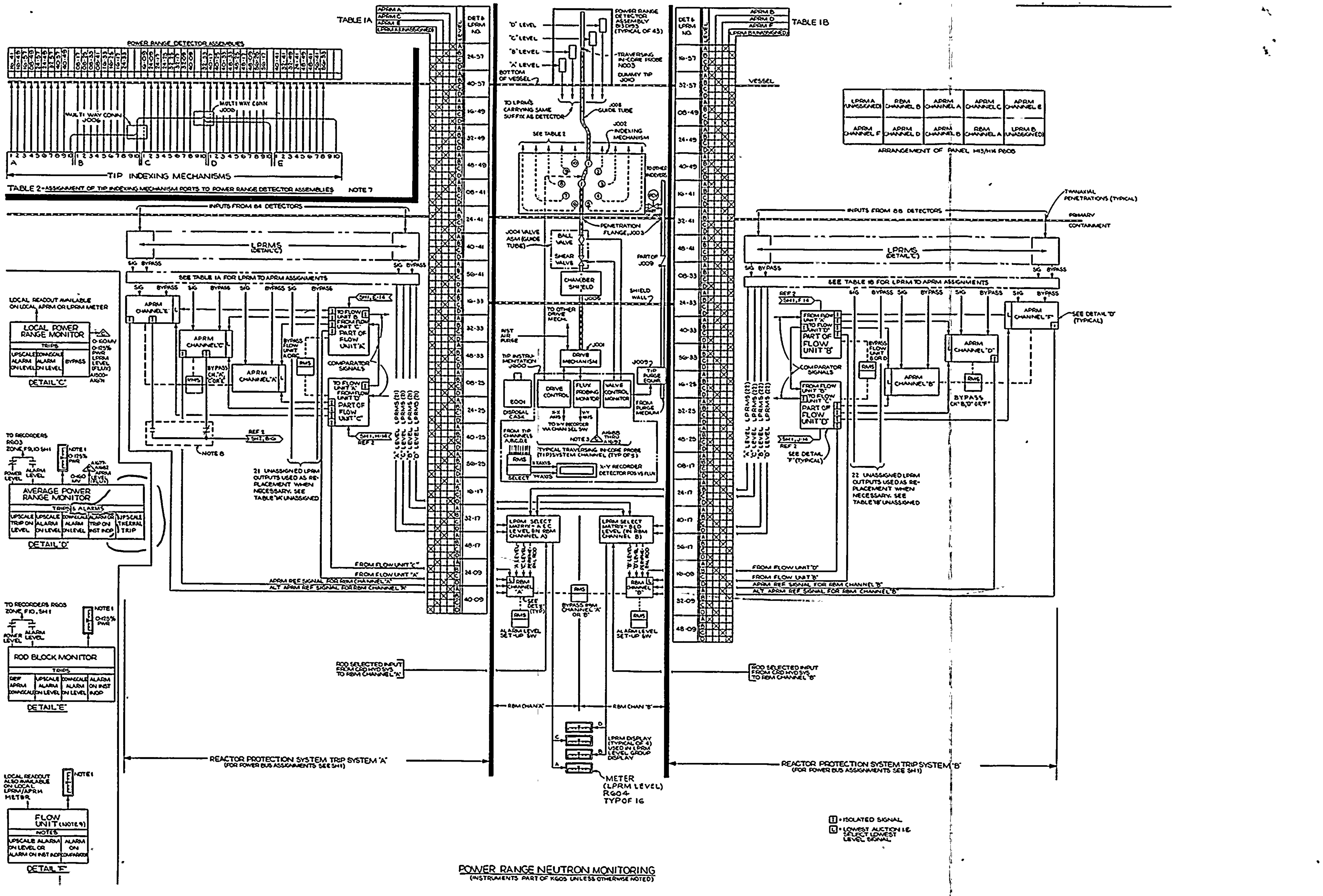
AMENDMENT NO. 16
June 1981

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 7.3- 9b

AMENDMENT NO. 16
June 1981

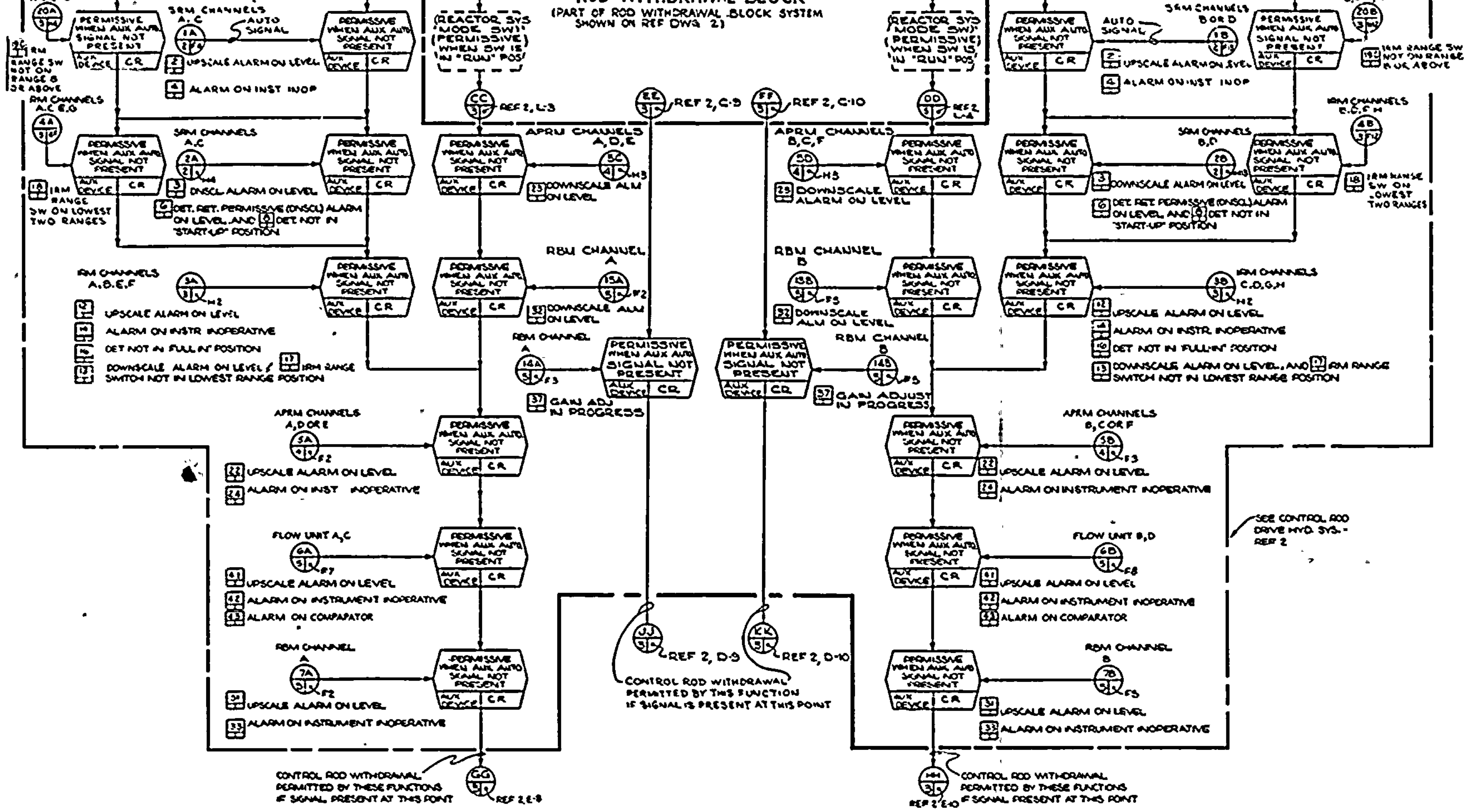
DELETED



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM IED - SHEET 2	FIGURE 7.6-4b

AMENDMENT NO. 16
June 1981

FIGURE 1
NEUTRON MONITORING SYSTEM
ROD WITHDRAWAL BLOCK
(PART OF ROD WITHDRAWAL BLOCK SYSTEM SHOWN ON REF DWG 2)



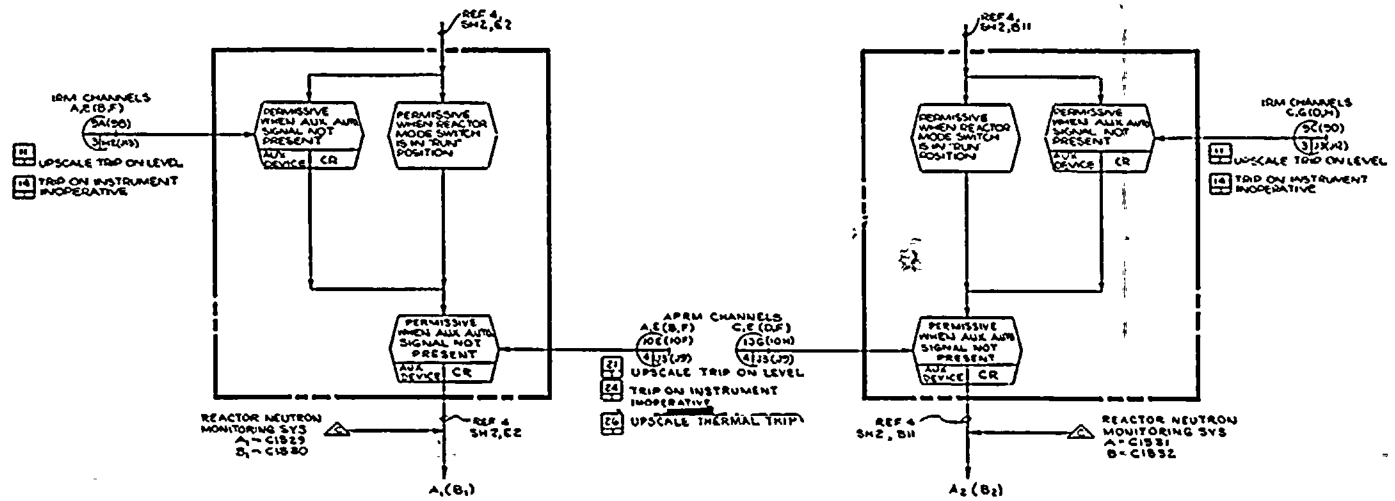
NOTES:
1. INPUTS TO COMPUTER ARE ISOLATED CLOSE TO ALARM CONTACTS.
2. SEE SHEET 7.
3. SEE SHEET 7.
4. THE ENTIRE NEUTRON MONITORING SYSTEM IS A FULLY AUTOMATIC SYSTEM EXCEPT FOR MANUAL OPERATED SWITCHES.
5. ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY C51 UNLESS OTHERWISE NOTED.
6. CHANNELS A,C,E & G ARE FOR TRIP SYSTEM A. CHANNELS B,D,F & H ARE FOR TRIP SYSTEM B.

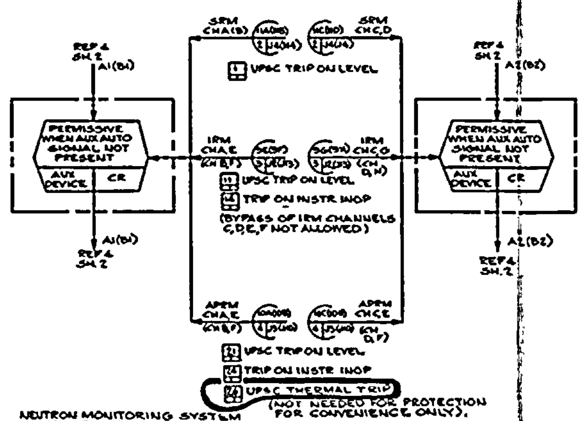
LEGEND:
IRM — INTERMEDIATE RANGE MONITOR
RBM — ROD BLOCK MONITOR
APRM — AVERAGE POWER RANGE MONITOR
SRM — SOURCE RANGE MONITOR
LPRM — LOCAL POWER RANGE MONITOR
TIP — TRAVERSING IN-CORE PROBE
MOC — MULTIPLE OUTPUT CONTROLLER

REFERENCE DOCUMENTS:
NOTE: SYSTEM SELECTION OPTIONS ARE INDICATED BY MULTIPLE MPL ITEM NUMBERS

	MPL ITEM NO.
1 NEUTRON MONITORING SYSTEM IED	C51-1010
2 CONTROL ROD DRIVE HYD SYS FCD	C11&C12-1030
3 NUCLEAR BOILER SYS FCD	B21-1030
4 REACTOR PROTECTION SYS IED	C71&C72-1010
5 PROCESS COMPUTER SYSTEM INPUT/OUTPUT REQUIREMENTS DESIGN SPEC	C91&C92-4010
6 LOGIC SYMBOLS	A41-1010
7 NEUTRON MONITORING SYS ARRANGEMENT	C51-2010

FIGURE 2
NEUTRON MONITORING SYSTEM TRIP - REACTOR PROTECTION SYSTEM TRIP SYSTEM A
(TYPICAL FOR TRIP SYSTEM B)





NEUTRON MONITORING SYSTEM NON-COINCIDENT TRIP (INITIAL LOADING ONLY) REACTOR PROTECTION SYSTEM TRIP SYSTEM A (TYPICAL FOR SYSTEM B)
FIGURE 3

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6a

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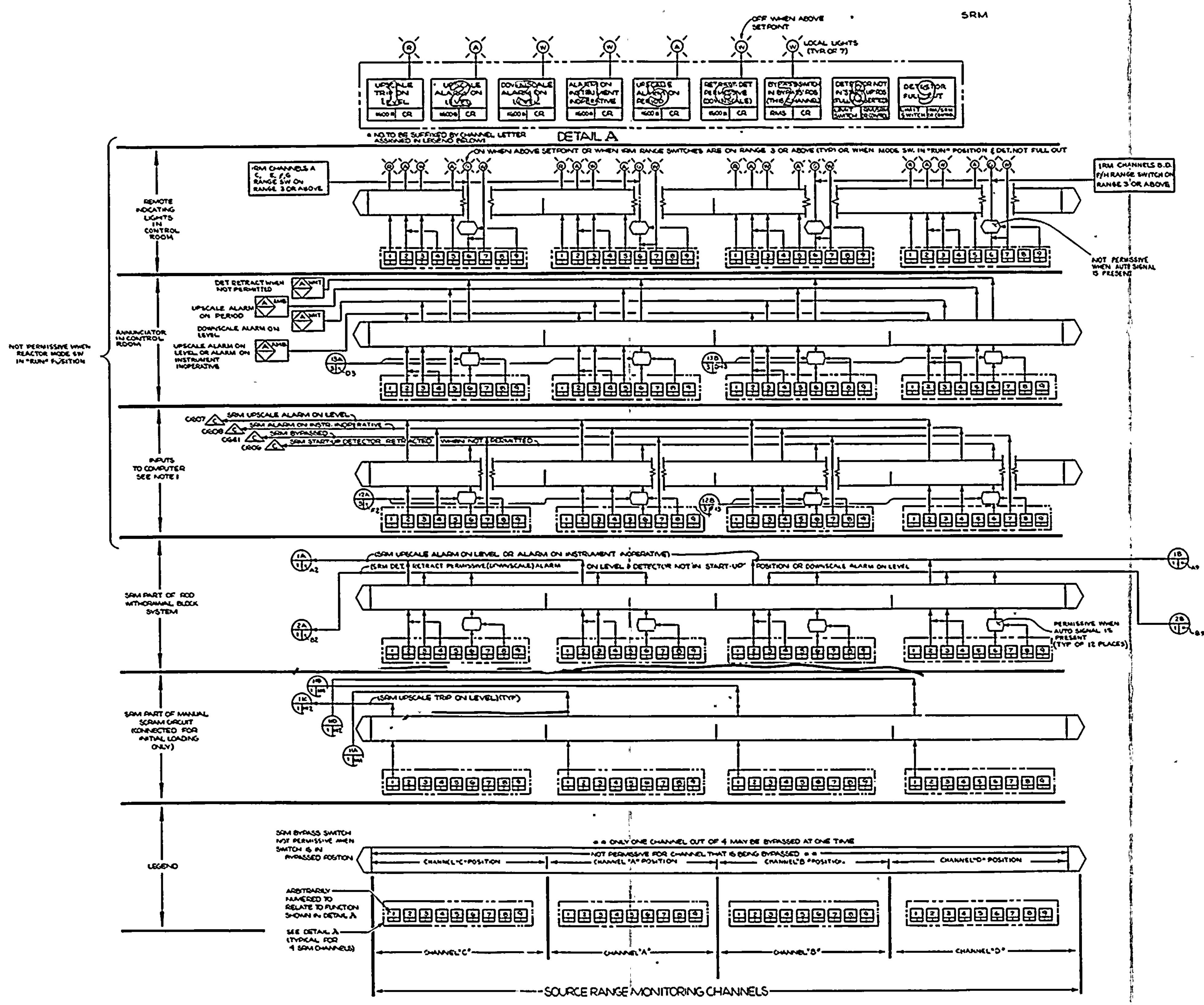

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM

FIGURE 7.6-6b

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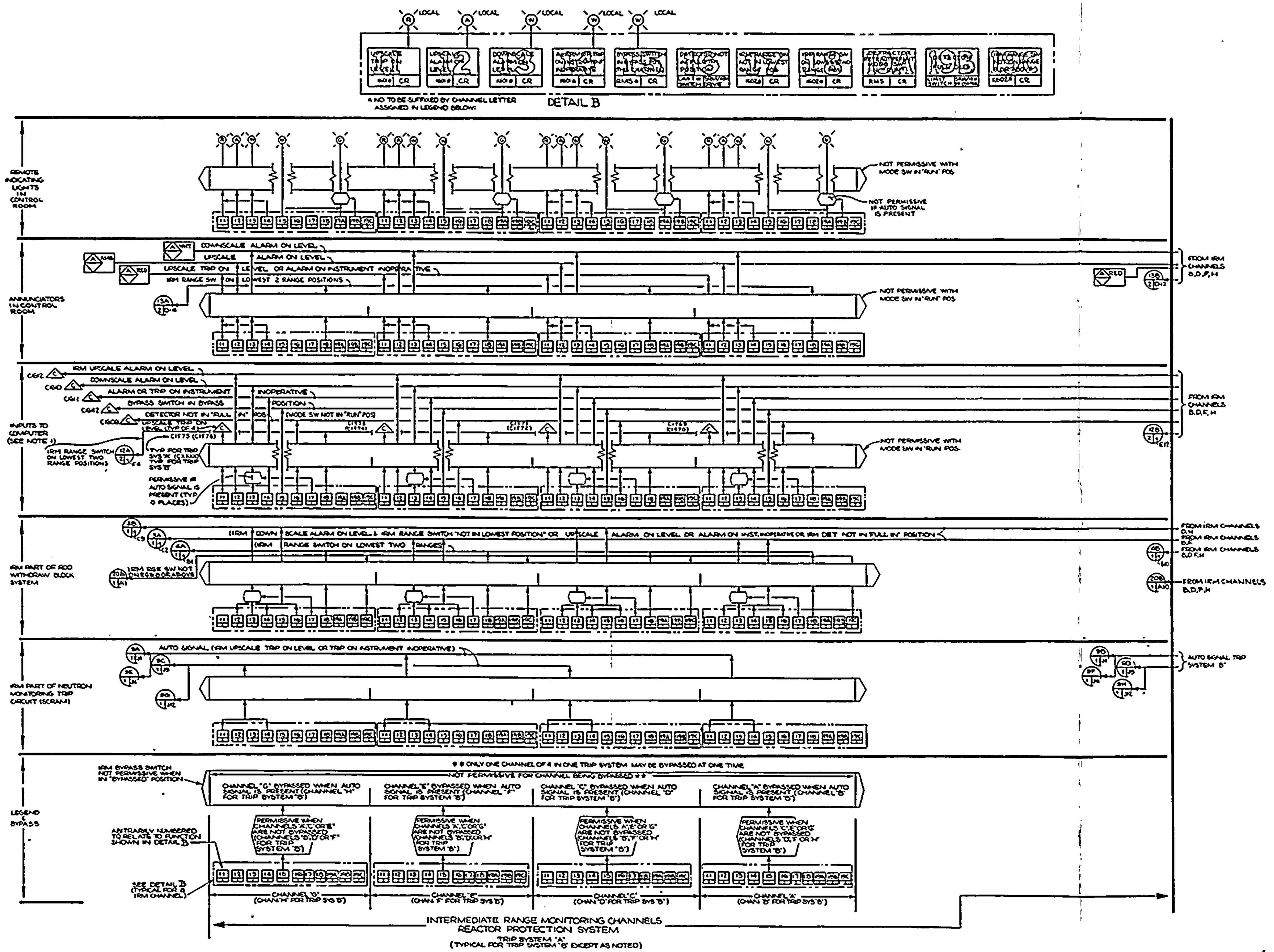

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6c

AMENDMENT NO. 16
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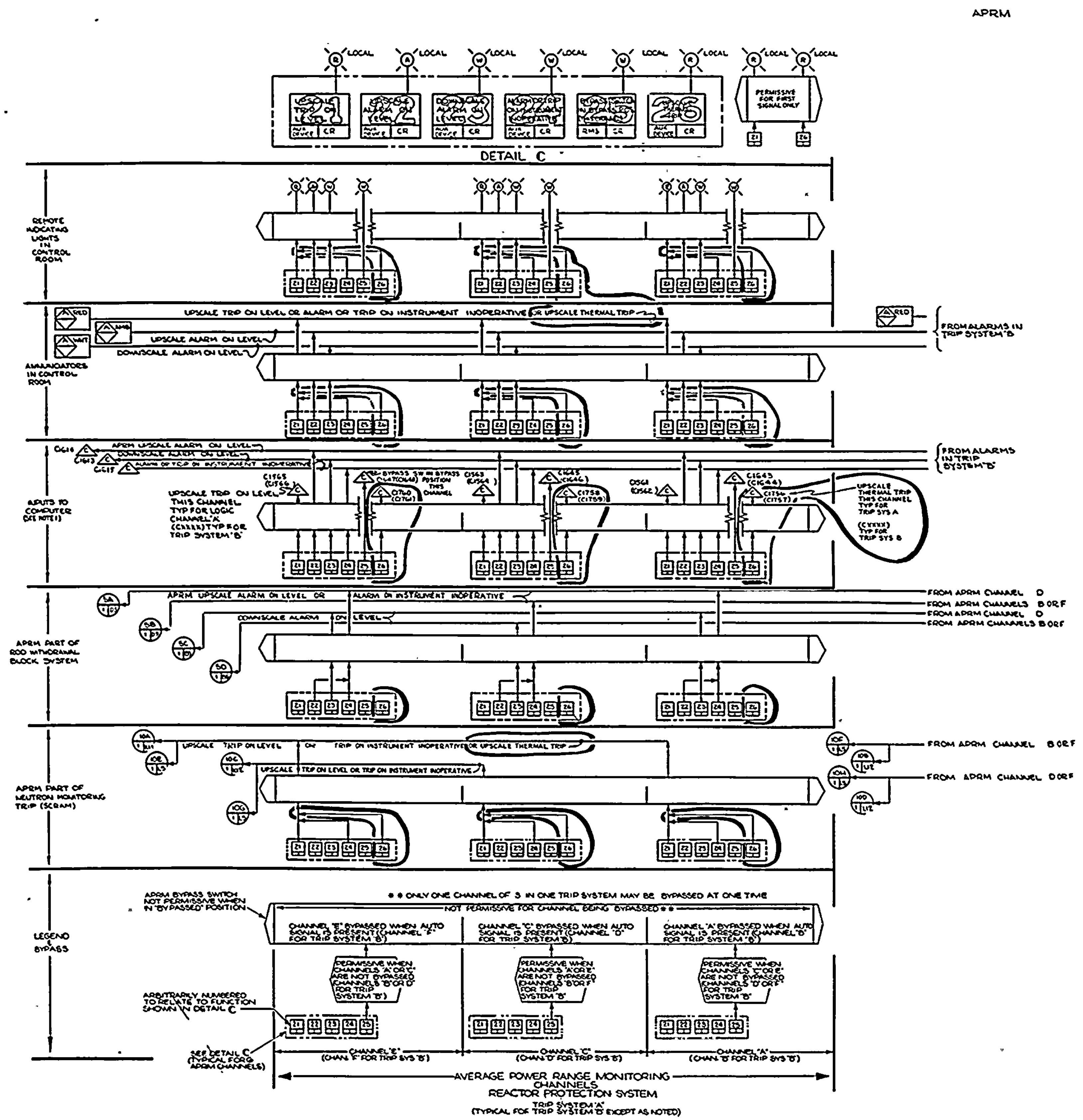

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM	FIGURE 7.6-6d

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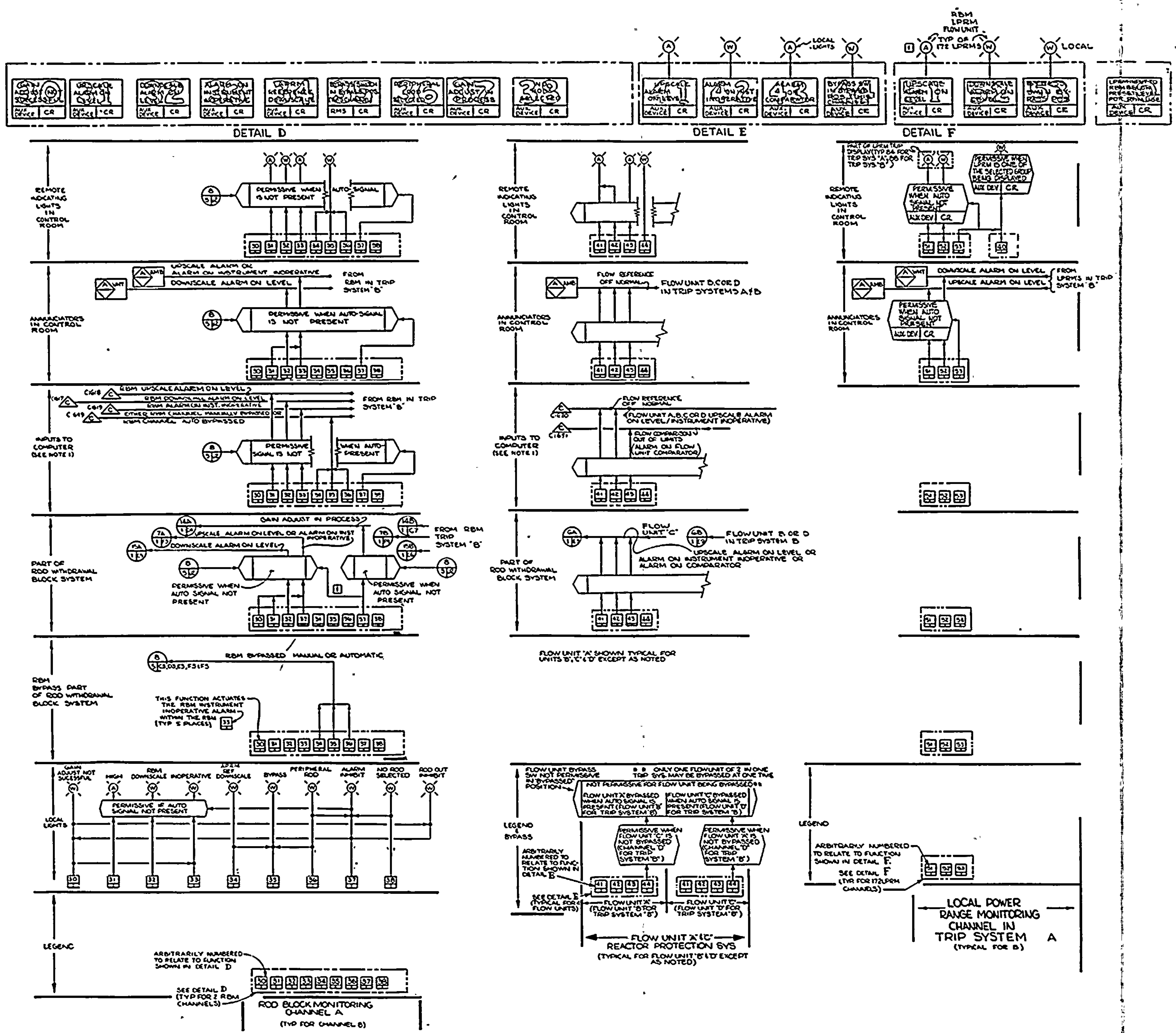

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONING CONTROL DIAGRAM	FIGURE 7.6-6e

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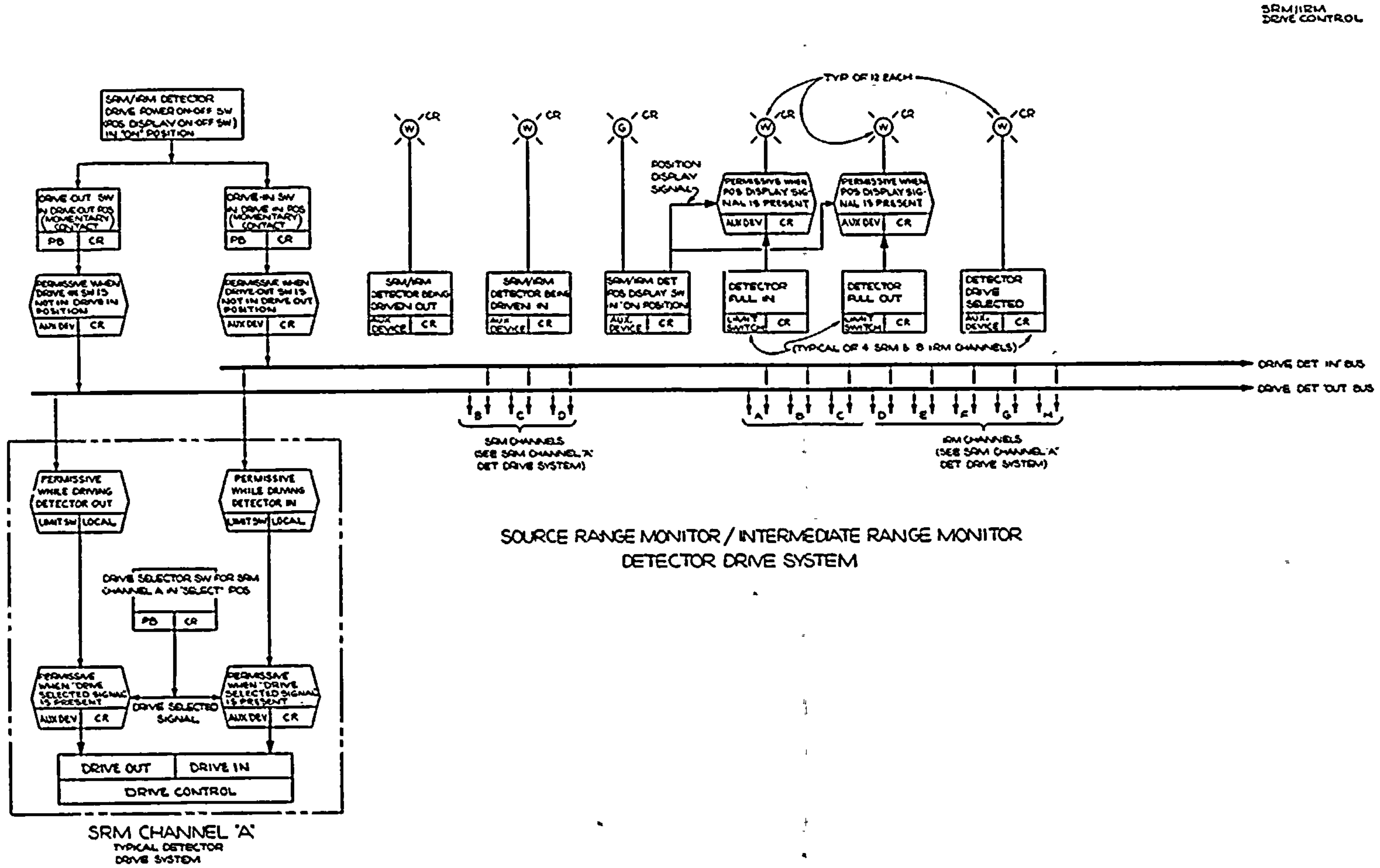

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM

FIGURE 7.6-6f

AMENDMENT NO. 16
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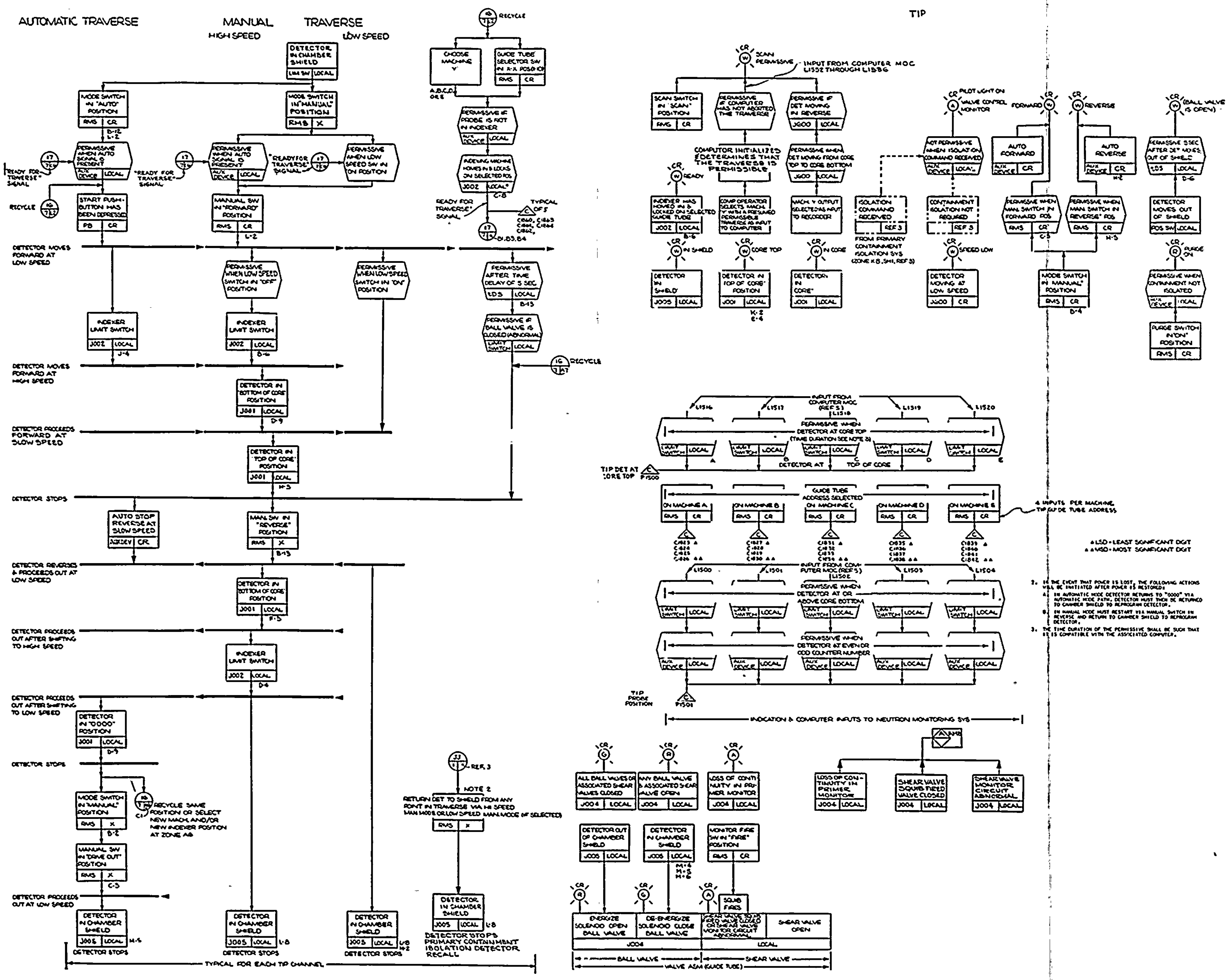
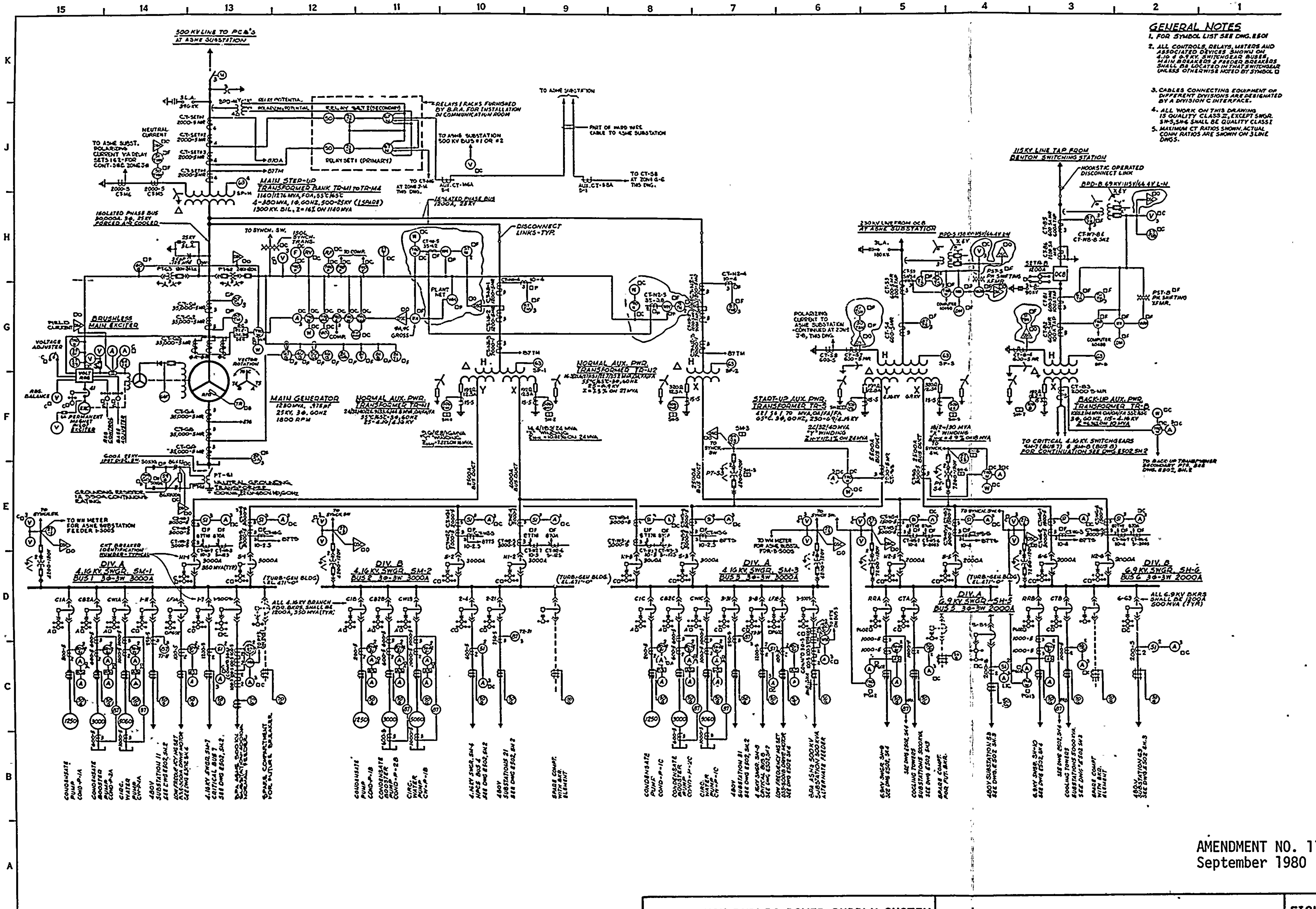


WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

NEUTRON MONITORING SYSTEM FUNCTIONAL CONTROL DIAGRAM

FIGURE 7.6-6g



AMENDMENT NO. 11
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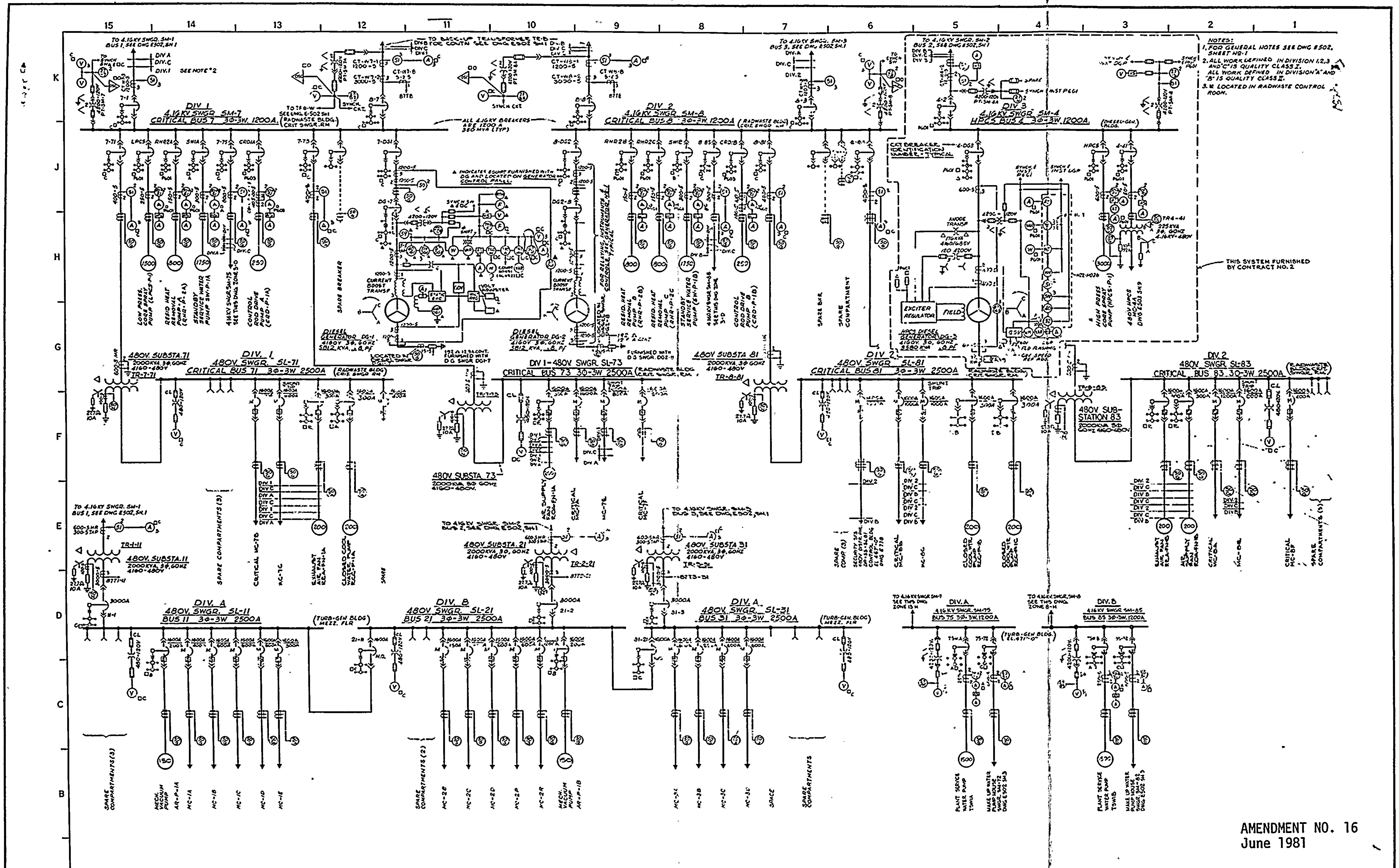
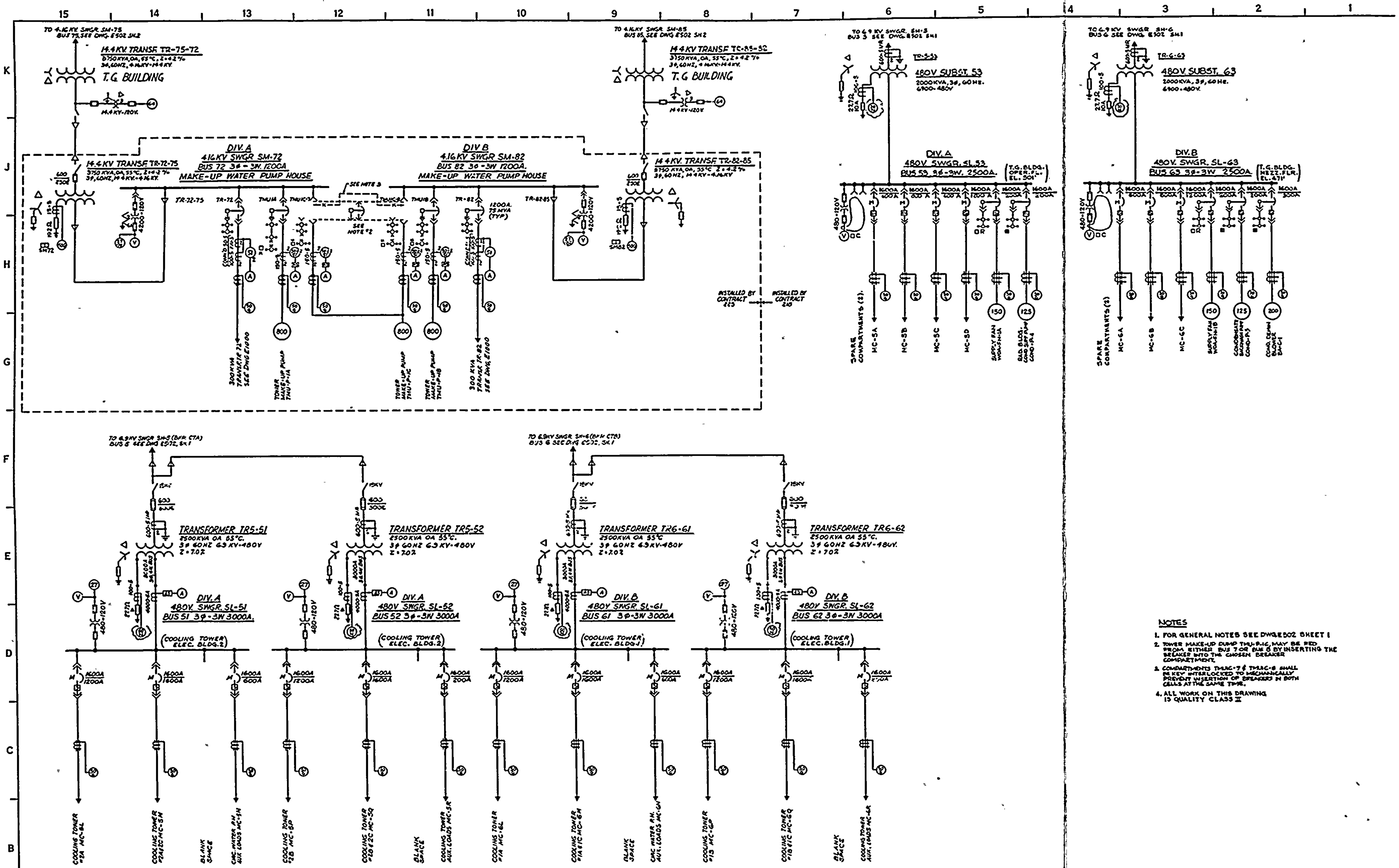
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN ONE LINE DIAGRAM - MAIN BUSES	FIGURE 8.1-9a



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN ONE LINE DIAGRAM - EMERGENCY BUSES	FIGURE 8.1-9b



NOTES

1. FOR GENERAL NOTES SEE DWG E502 SHEET I
2. TOWER MAKE-UP PUMP TMU-P-1C, MAY BE FED FROM EITHER BUS 7 OR BUS 8 BY INSERTING THE BREAKER INTO THE CHOSEN BREAKER COMPARTMENT.
3. COMPARTMENTS TMUC-7 & TMUC-8 SHALL BE KEY INTERLOCKED TO MECHANICALLY PREVENT INSERTION OF BREAKERS IN BOTH CELLS AT THE SAME TIME.
4. ALL WORK ON THIS DRAWING IS QUALITY CLASS II

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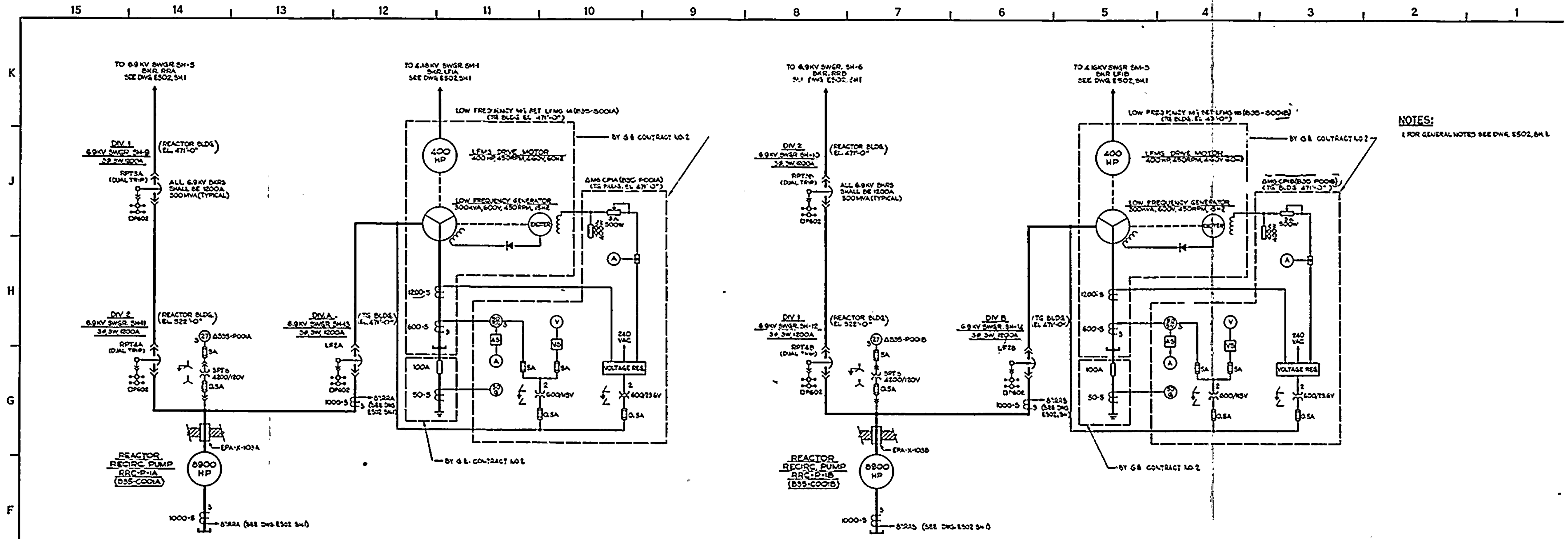
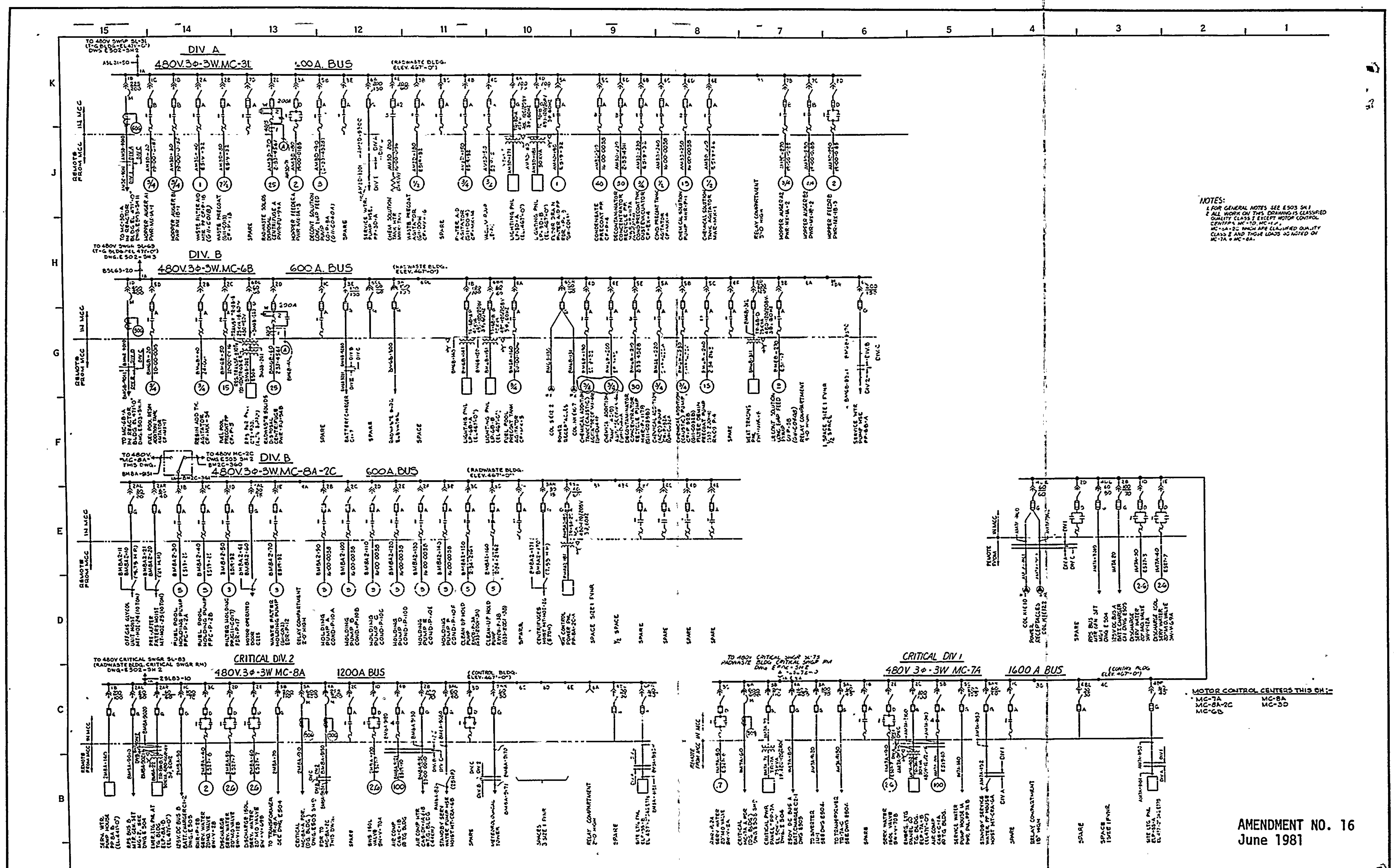
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN ONE LINE DIAGRAM - MISCELLANEOUS BUSES	FIGURE 8.1-9c



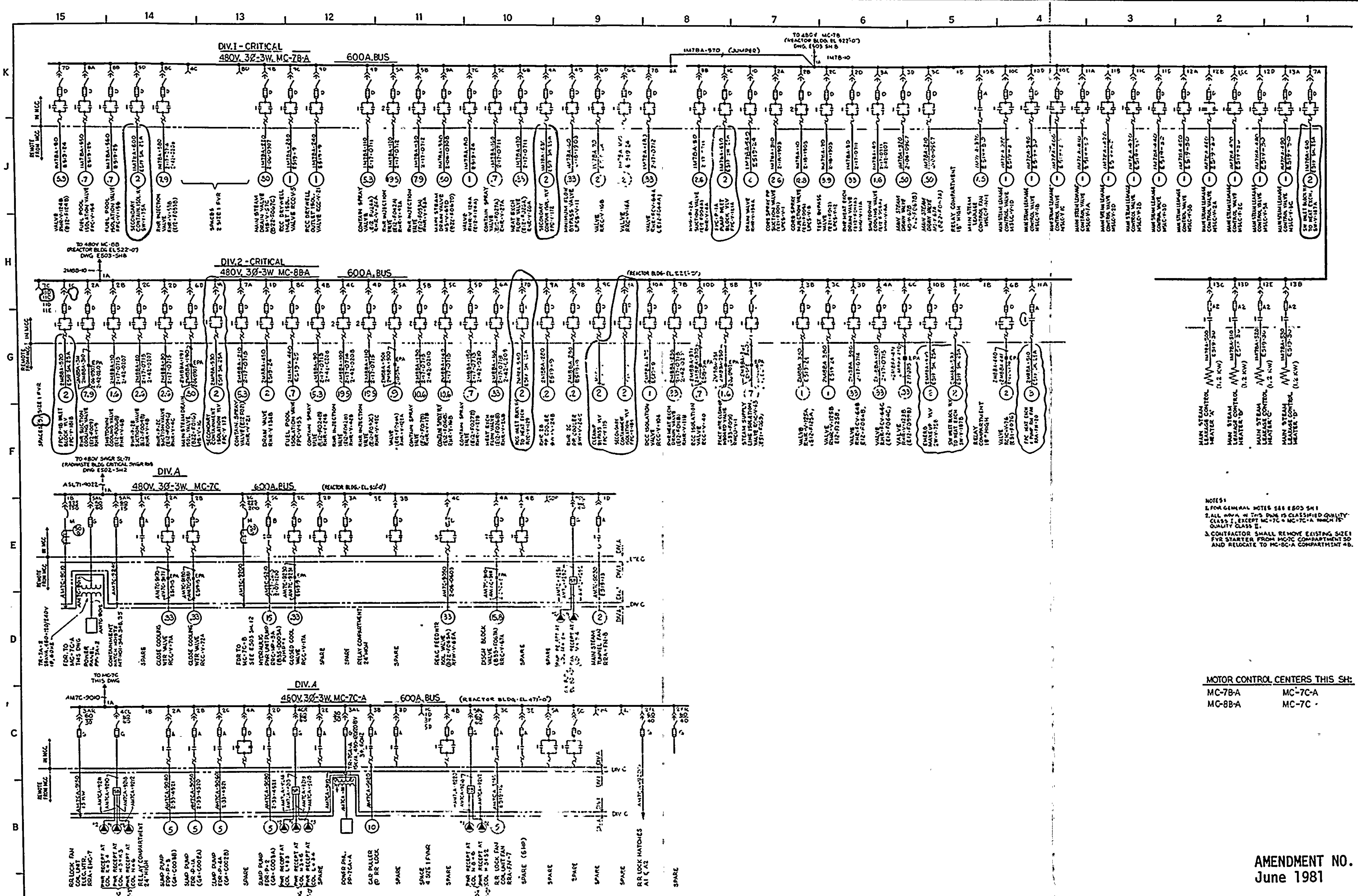
Amendment No. 7
November 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CABLE TRAY PLANS RADWASTE & CONTROL BLDG. CABLE SPREADING ROOM	FIGURE 8.1-9d



AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AUXILIARY ONE LINE DIAGRAM (MCC's)	FIGURE 8.3-1a



NOTES:

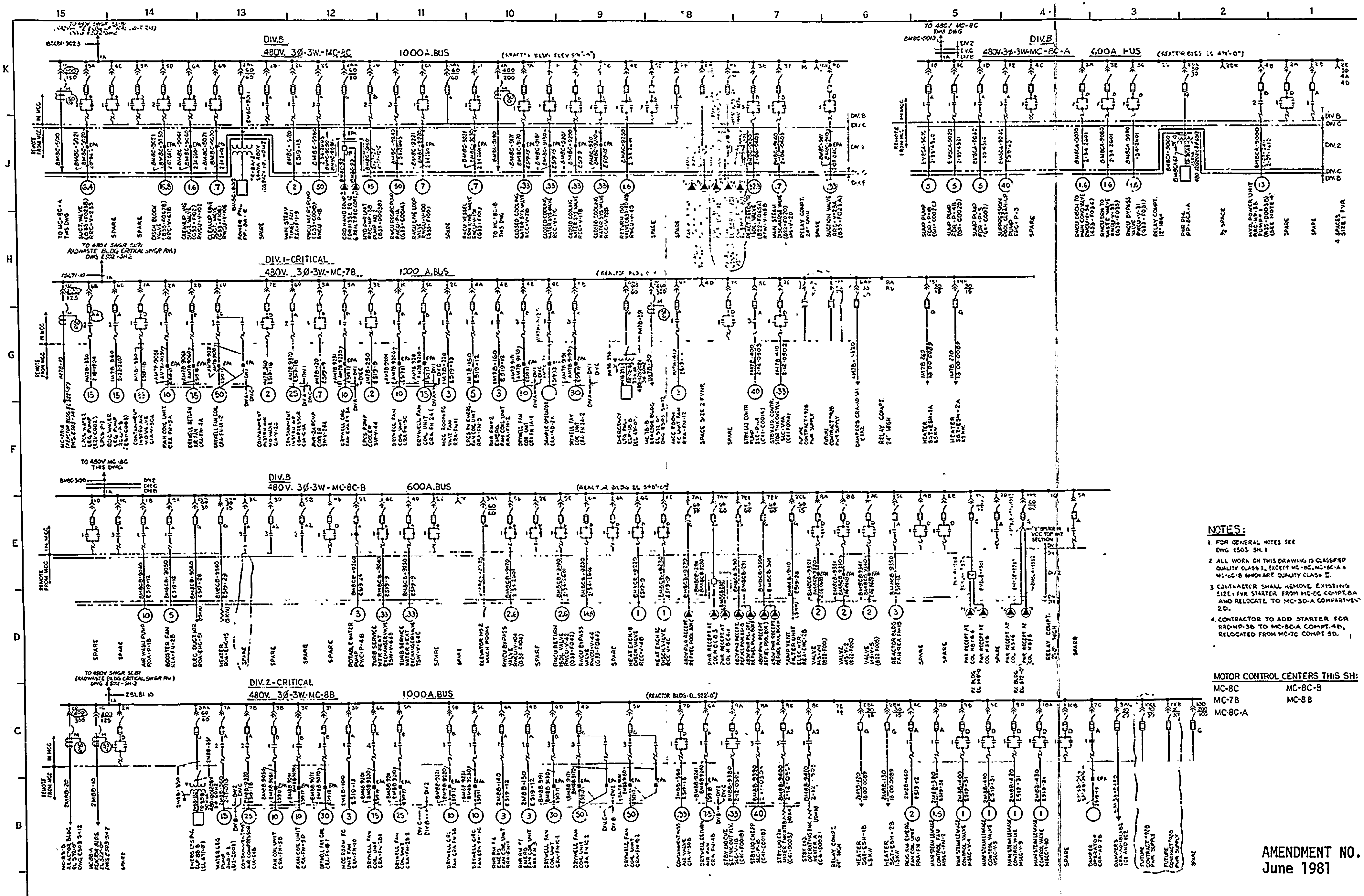
1. FOR GENERAL NOTES SEE E503 SH 1
2. ALL WORK IN THIS DWG IS CLASSIFIED QUALITY CLASS I, EXCEPT MC-7C & MC-7C-A WHICH IS QUALITY CLASS II.
3. CONTRACTOR SHALL REMOVE EXISTING SIZE 1 FVR STARTER FROM MC-7C COMPARTMENT 5D AND RELOCATE TO MC-8C-A COMPARTMENT 4D.

MOTOR CONTROL CENTERS THIS SH:

MC-7B-A	MC-7C-A
MC-8B-A	MC-7C

AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AUXILIARY ONE LINE DIAGRAM (MCC'S)	FIGURE 8.3-1b



NOTES:

1. FOR GENERAL NOTES SEE DWG E503 SH. 1
2. ALL WORK ON THIS DRAWING IS CLASSIFIED QUALITY CLASS I, EXCEPT MC-8C, MC-8C-A & MC-8C-B WHICH ARE QUALITY CLASS II.
3. CONTRACTOR SHALL REMOVE EXISTING SIZE 1 FVR STARTER FROM MC-8C COMPT. 8A AND RELOCATE TO MC-3D-A COMPARTMENT 2D.
4. CONTRACTOR TO ADD STARTER FOR RRC-HP-3B TO MC-8C-A COMPT. 4D, RELOCATED FROM MC-7C COMPT. 5D.

MOTOR CONTROL CENTERS THIS SH:

MC-8C	MC-8C-B
MC-7B	MC-8B
MC-8C-A	

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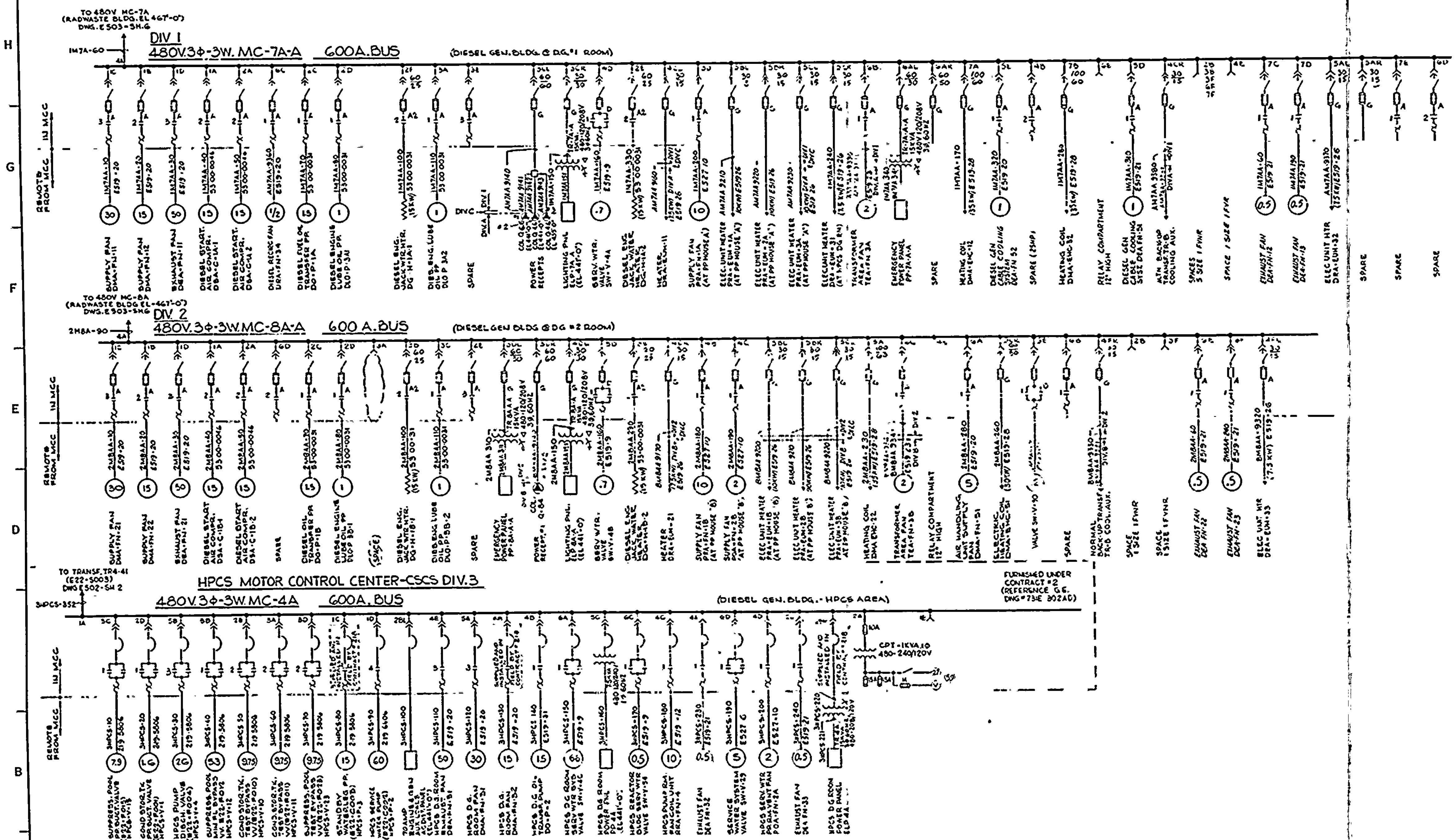
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AUXILIARY ONE LINE DIAGRAM (MCC'S)	FIGURE 8.3-1c

NOTES:

1. FOR GENERAL NOTES SEE DWG. E503 SH.1
2. ALL WORK ON THIS DRAWING IS CLASSIFIED QUALITY CLASS I



MOTOR CONTROL CENTERS THIS SH:- MC-7A-A MC-8A-A MC-4A

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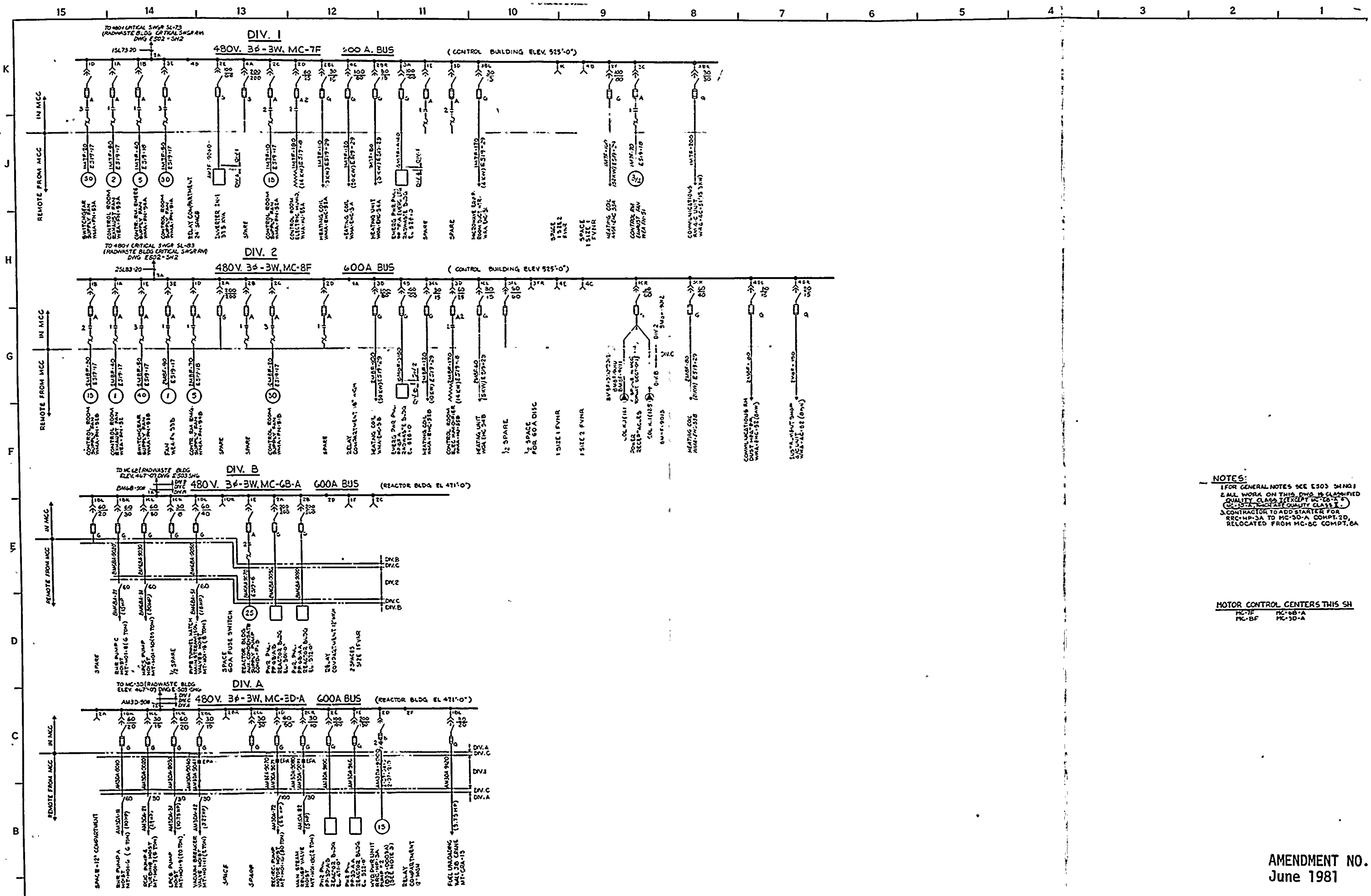
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AUXILIARY ONE LINE DIAGRAM (MCC'S)	FIGURE 8.3-1d



NOTES:
1. FOR GENERAL NOTES SEE E503 SH NO. 1
2. ALL WORK ON THIS DWG IS CLASSIFIED QUALITY CLASS I (EXCEPT MC-6B-A & MC-3D-A, WHICH ARE QUALITY CLASS II.)
3. CONTRACTOR TO ADD STARTER FOR RRC-HP-3A TO MC-3D-A COMPT. 2D, RELOCATED FROM MC-8C COMPT. 6A

MOTOR CONTROL CENTERS THIS SH

MC-7F	MC-6B-A
MC-8F	MC-3D-A

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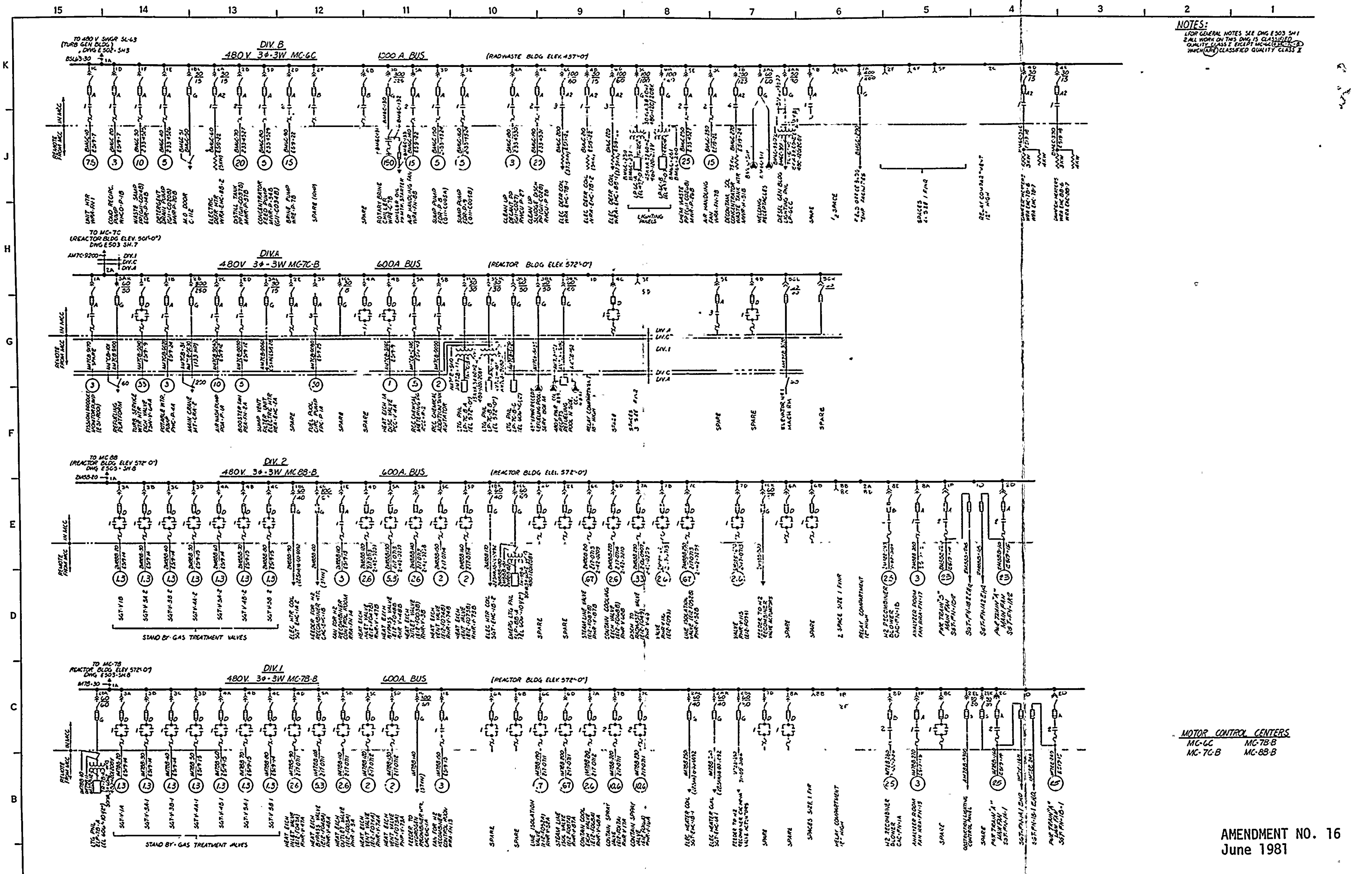
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	AUXILIARY ONE LINE DIAGRAM (MCC'S)	FIGURE 8.3-1e

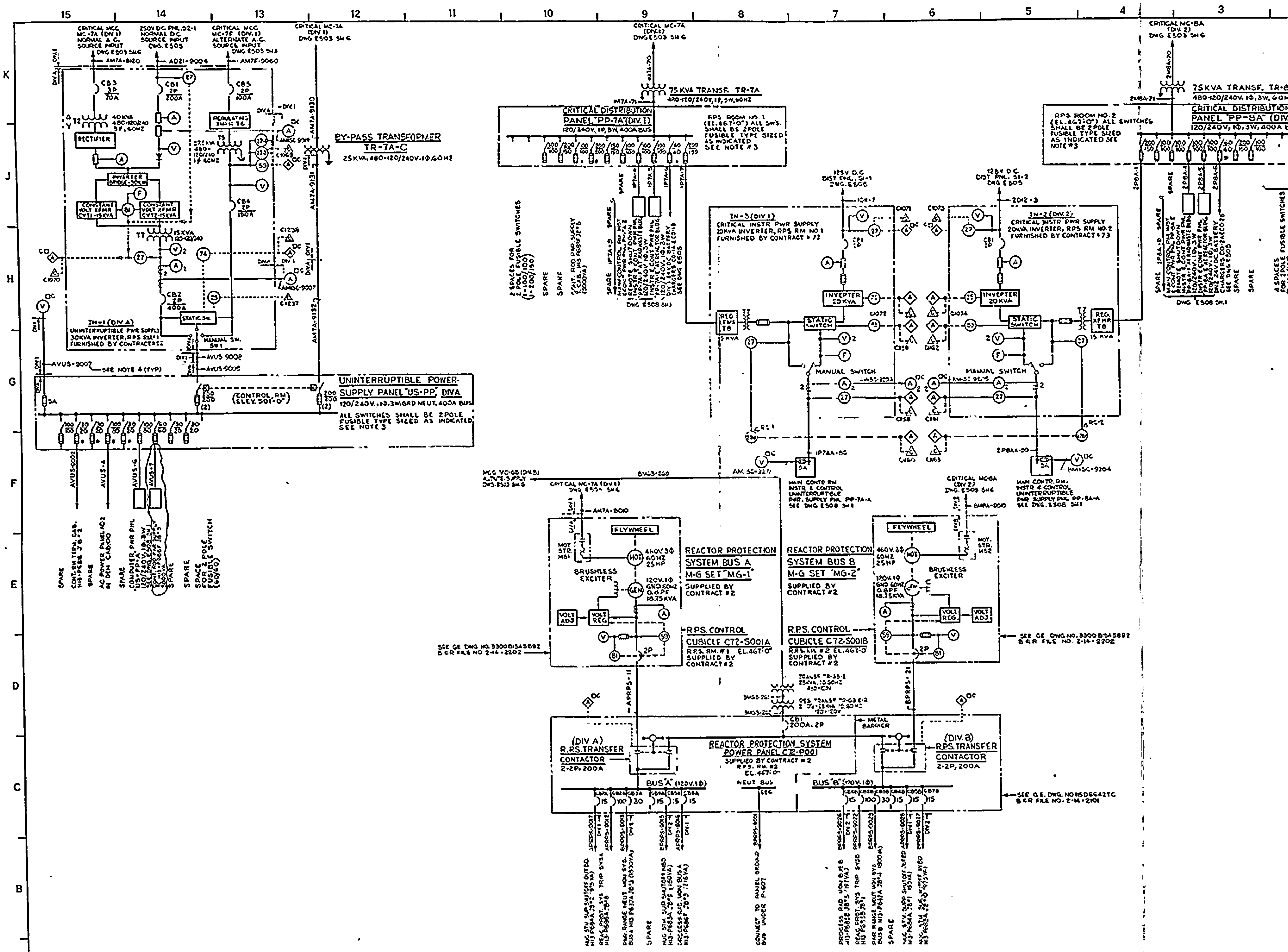


AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

AUXILIARY ONE LINE DIAGRAM (MCC's)

FIGURE 8.3-1f



NOTES:

1. WORK ON THIS DWG. TO BE CLASSIFIED, QUALITY CLASS I, SEISMIC CLASS I, EXCEPT EQUIPMENT ASSOCIATED WITH THE "UPS" WHICH IS QUALITY CLASS II, SEISMIC CLASS II AND ARE ENCLOSED WITHIN A SEISMIC CLASS I ENCLOSURE.
2. ALL EQUIPMENT SHALL BE FURNISHED AND INSTALLED BY CONTRACT #219 UNLESS SPECIFIED OTHERWISE.
3. TWO POLE SWITCHES INDICATED BY * SHALL UTILIZE ONE FUSE (SIZE AS INDICATED) AND ONE POLE ONLY. THE REMAING POLE AND FUSE POSITION TO REMAINING SPARE. SWITCHES USED AS ONE POLE ONLY SHALL BE CONNECTED TO PANEL BUSES TO OBTAIN BALANCED LOADS BETWEEN PANEL BUSES.
4. CABLES DESIGNATED IN XXXX-9XXX SERIES NUMBERS ARE CLASSIFIED AS "ASSOCIATED CIRCUITS". THESE CABLES MAY HAVE TWO CABLE DESIGNATIONS FOR ROUTING PURPOSES. THESE CABLES SHALL BE CONTINUOUS FROM ONE PIECE OF EQUIPMENT TO ANOTHER. THERE SHALL NOT BE ANY SPLICES OR TERMINATIONS AT TRANSITION POINTS, FIRE STOPS OR CABLE NUMBER IDENTIFICATION CHANGES.

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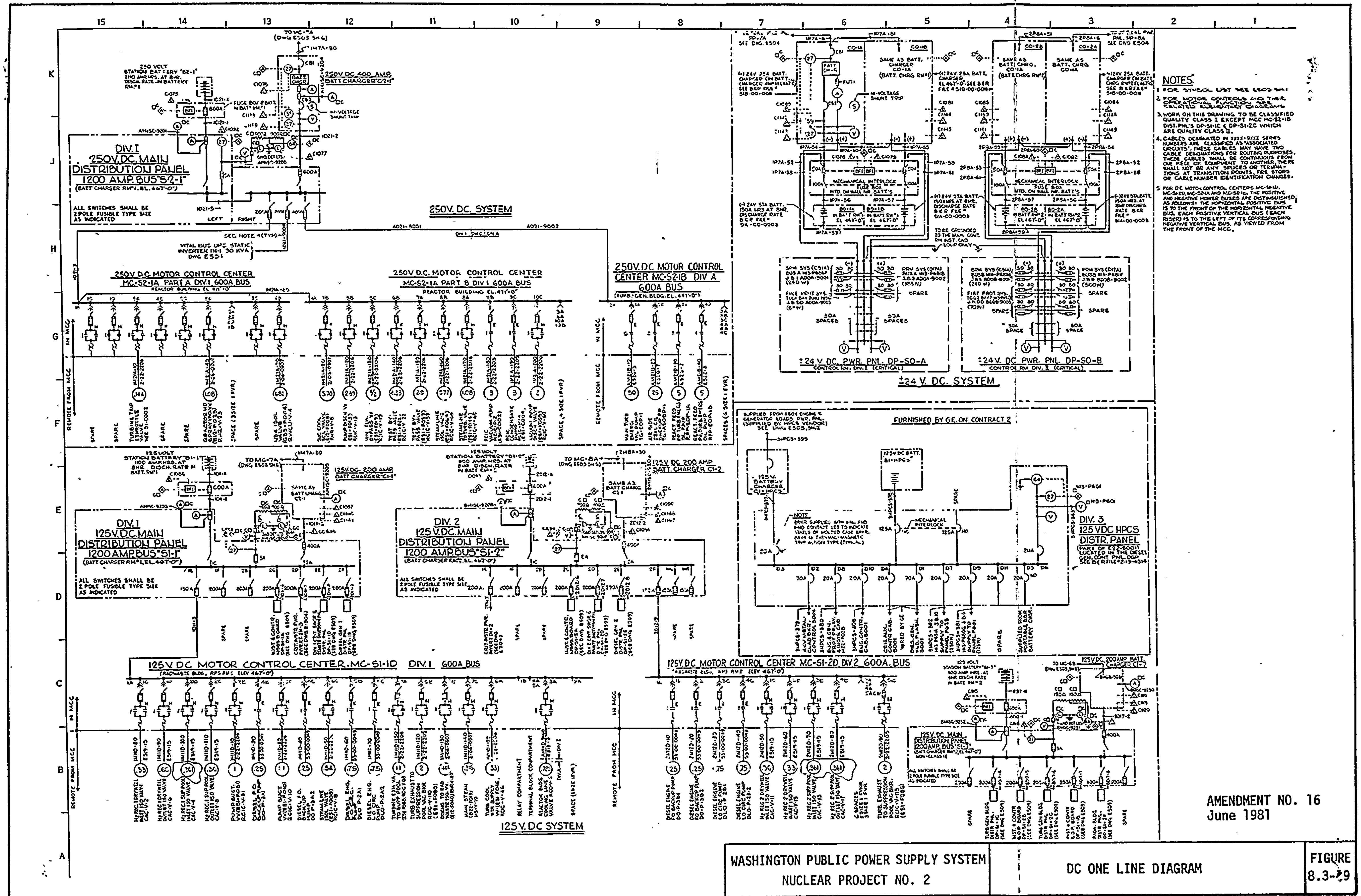
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

VITAL ONE LINE DIAGRAM

FIGURE 8.3-2



DIV.I 250V.DC.MAIN DISTRIBUTION PANEL 1200 AMP.BUS"S2-1"
250V DC 400 AMP BATT CHARGER C2-1
250V. DC. SYSTEM
250V D.C. MOTOR CONTROL CENTER MC-S2-1A PART A DIV.I 600A BUS
250V D.C. MOTOR CONTROL CENTER MC-S2-1A PART B DIV I 600A BUS
250V.DC MOTOR CONTROL CENTER MC-S2-1B DIV A 600A BUS
±24 V. DC. PWR. PNL. DP-SO-A
±24 V. DC. PWR. PNL. DP-SO-B
±24 V. DC. SYSTEM
NOTES:
1. FOR SYMBOL LIST SEE E503 SH.1
3. WORK ON THIS DRAWING TO BE CLASSIFIED QUALITY CLASS I EXCEPT MCC MC-S2-1B DIST.PNL'S DP-S1-1C & DP-S1-2C WHICH ARE QUALITY CLASS II.
FURNISHED BY G.E. ON CONTRACT 2
DIV.I 125V.DC.MAIN DISTRIBUTION PANEL 1200 AMP BUS "S1-1"
125V.DC. 200 AMP BATT CHARGER "C1-1"
DIV. 2 125V.DC.MAIN DISTRIBUTION PANEL 1200 AMP BUS "S1-2"
125V. DC 200 AMP BATT. CHARGER C1-2
DIV. 3 125VDC HPCS DISTR. PANEL
125V DC MOTOR CONTROL CENTER MC-S1-1D DIV.I 600A BUS
125V DC MOTOR CONTROL CENTER MC-S1-2D DIV 2 600A BUS
125V.DC SYSTEM
AMENDMENT NO. 16
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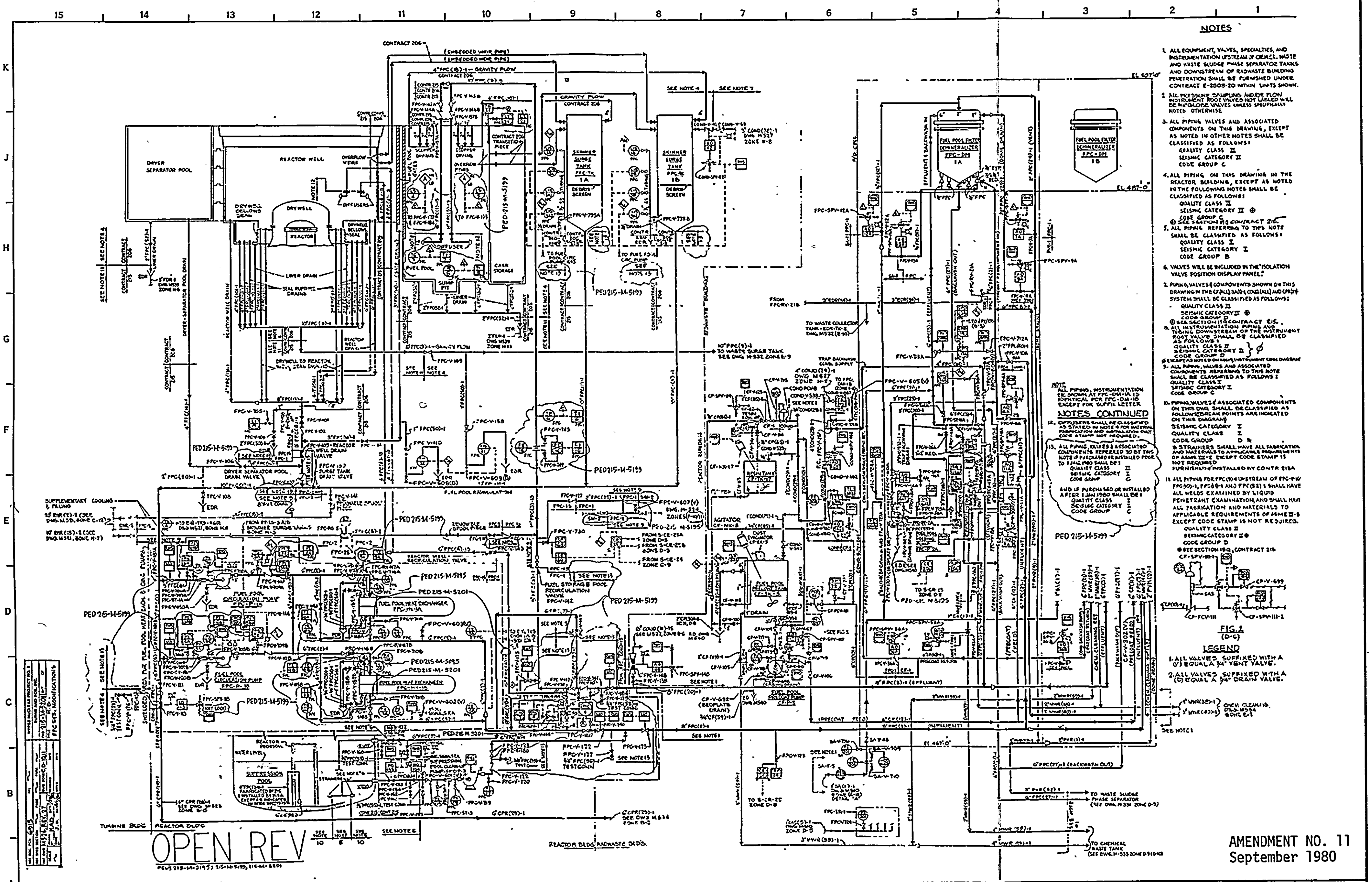
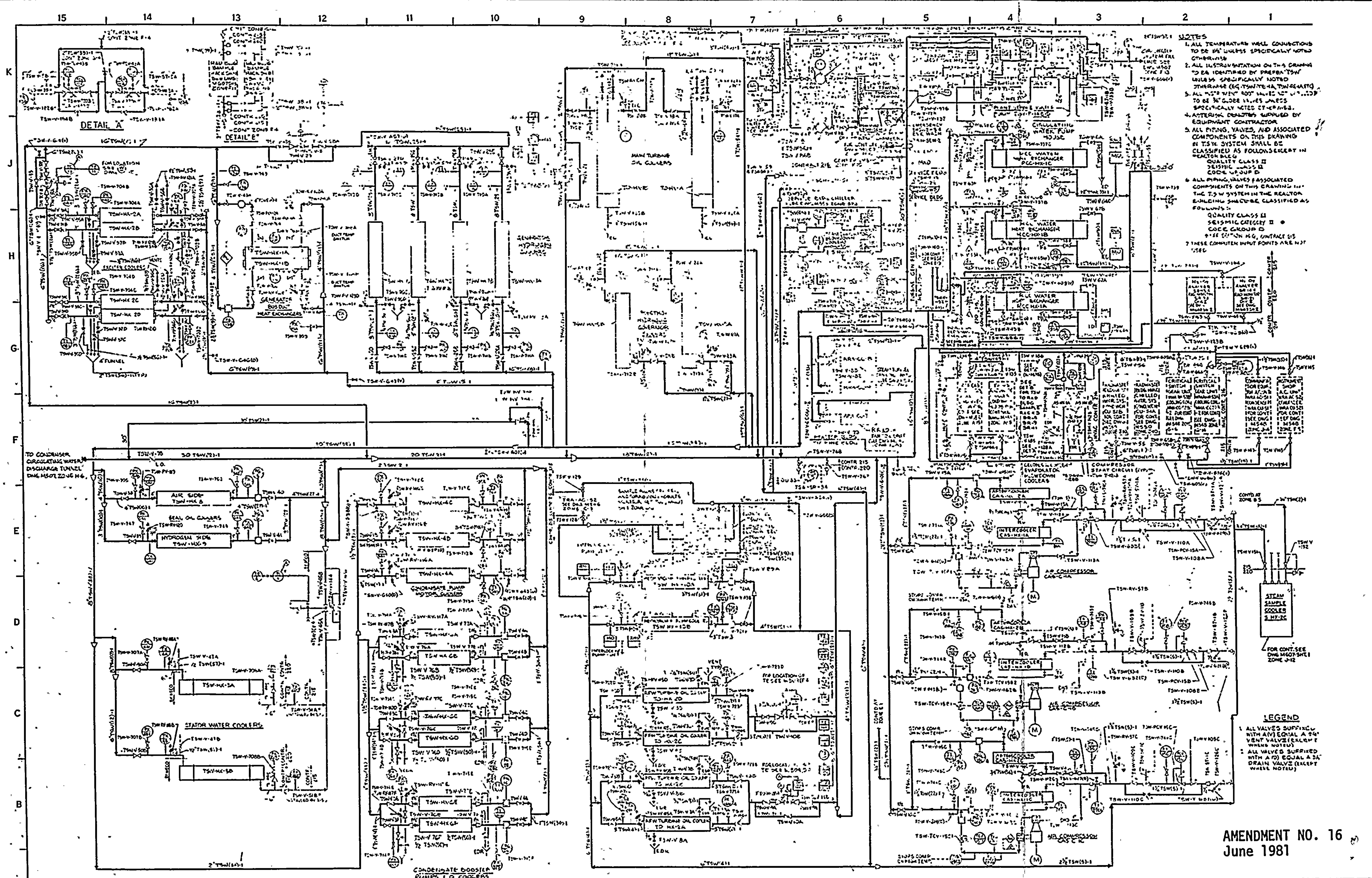
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2
DC ONE LINE DIAGRAM
FIGURE 8.3-29



AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FUEL POOL COOLING AND CLEANUP SYSTEM	FIGURE 9.1-4



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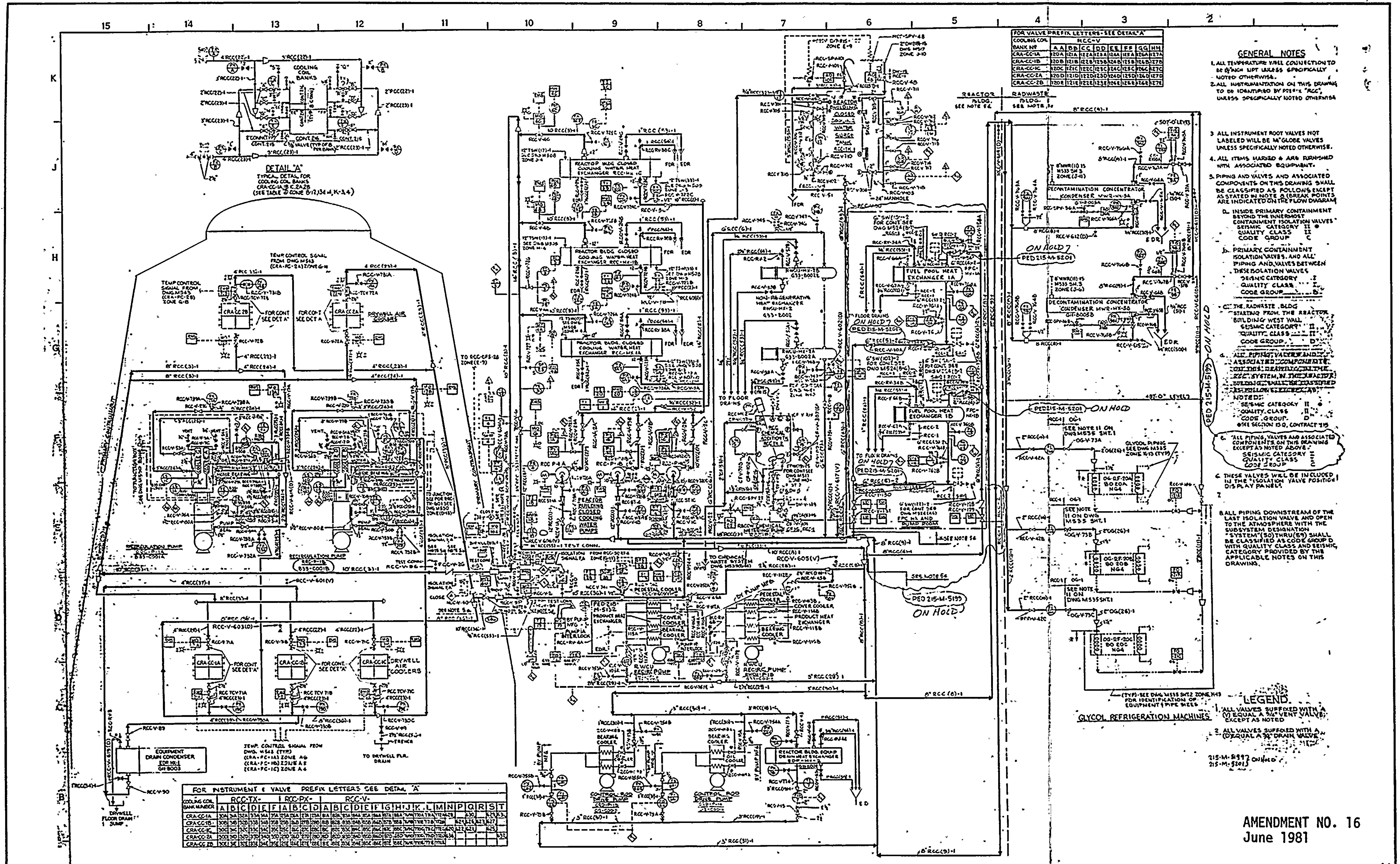
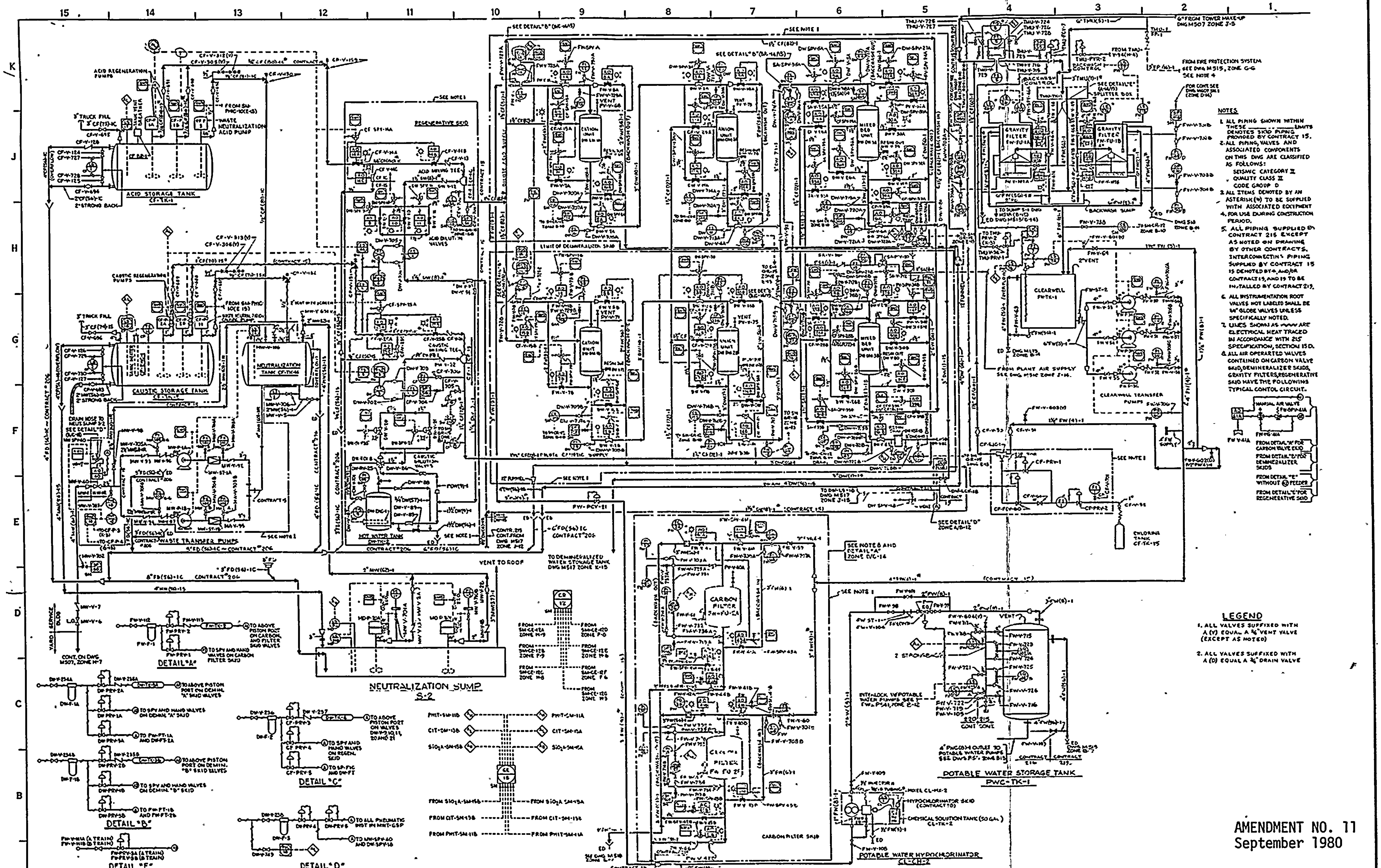
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PLANT SERVICE WATER SYSTEM	FIGURE 9.2-1



AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CLOSED COOLING WATER SYSTEM REACTOR AND RADWASTE BUILDING	FIGURE 9.2-2



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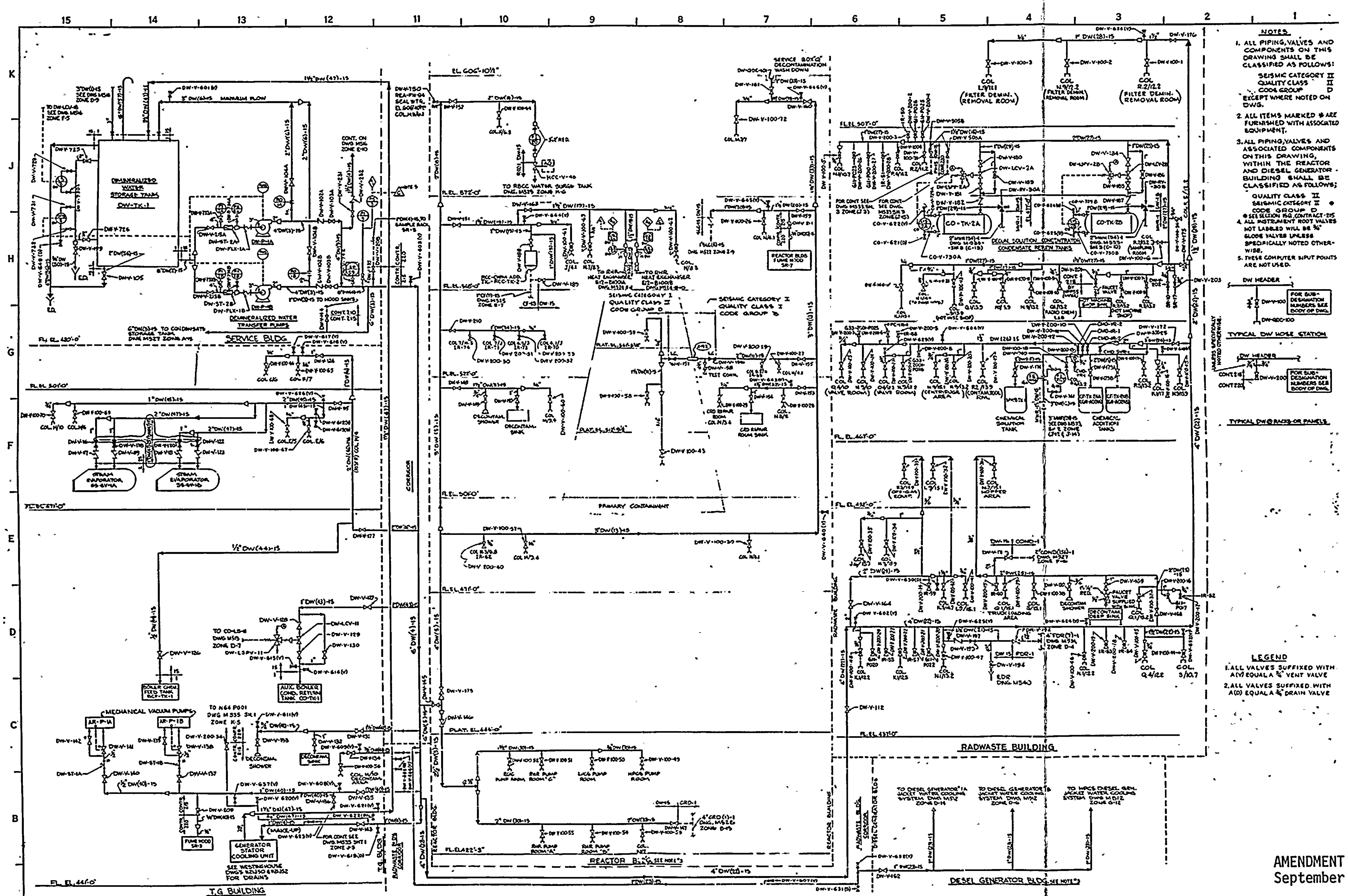
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PLANT MAKEUP WATER TREATMENT SYSTEM	FIGURE 9.2-3



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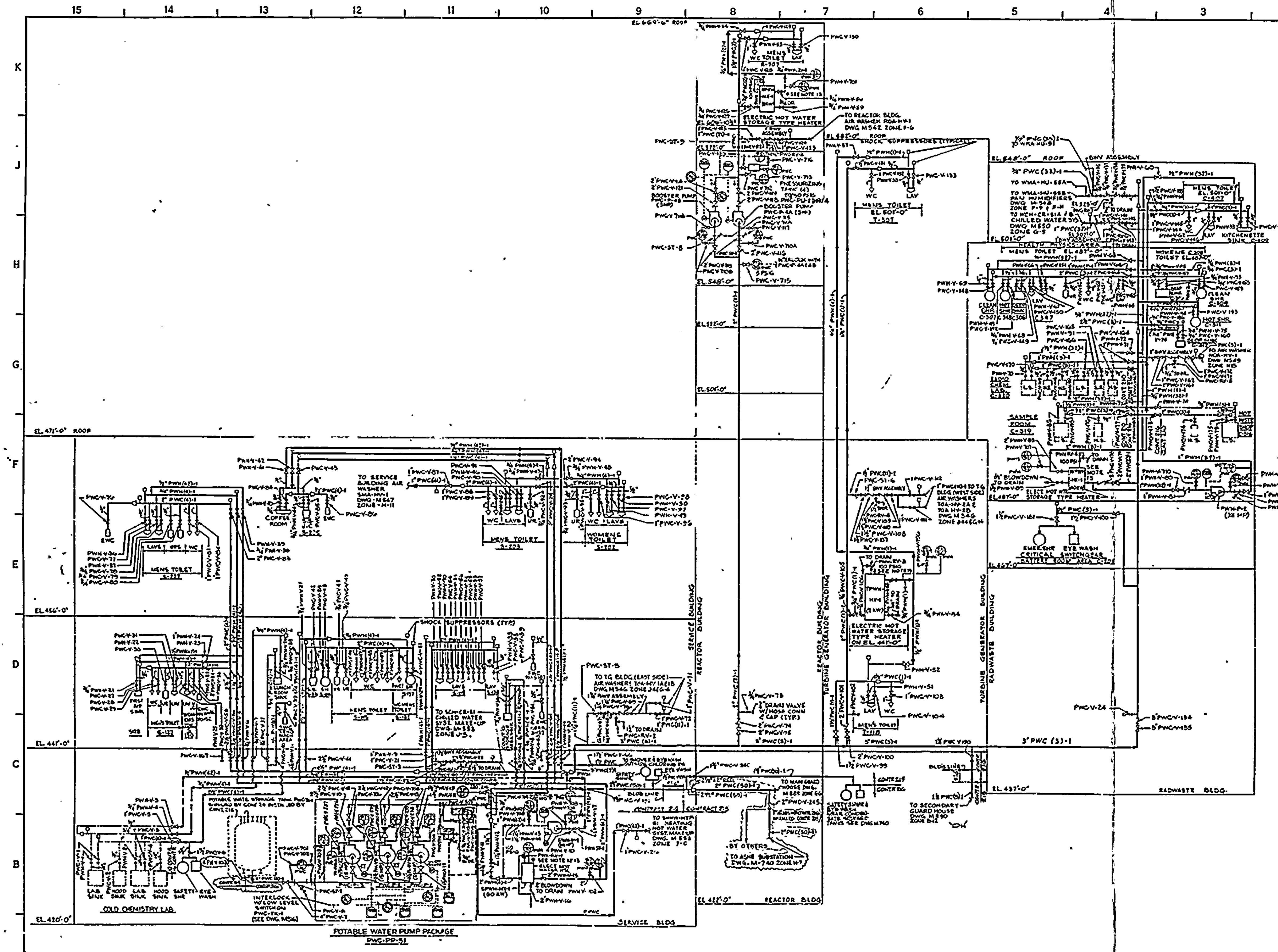
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	DEMINERALIZED WATER SYSTEM	FIGURE 9.2-4



NOTES:

1. USE ½" BRANCH SIZE FOR ALL LAVATORIES.
2. USE ¾" BRANCH SIZE FOR ALL URINALS.
3. USE 1" BRANCH SIZE FOR ALL WATER CLOSETS.
4. USE ¾" BRANCH SIZE FOR ALL SERVICE SINKS.
5. USE ½" BRANCH SIZE FOR ALL SHOWER HEADS.
6. USE ½" BRANCH SIZE FOR ALL ELECTRIC WATER COOLERS.
7. USE ½" BRANCH SIZE FOR ALL LAB SINKS.
8. SHOCK SUPPRESSORS SHALL BE PROVIDED FOR ALL HOT & COLD WATER SUPPLY BRANCHS AS SHOWN ON DIAGRAM BY SYMBOL
9. ALL VALVES IN PWC & PWH SYSTEM SHALL BE ½" SIZE UNLESS OTHERWISE NOTED
10. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG M501
11. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE ¾" GLOBE VALVES UNLESS NOTED OTHERWISE.
12. ALL PIPING VALVES AND ASSOCIATED EQUIPMENT ON THIS DWG SHALL BE CLASSIFIED AS SEISMIC CATEGORY II, QUALITY CLASS "Q", EXCEPT AS NOTED BELOW

    ALL PIPING AND PIPING SUPPORT IN REACTOR BLDG. & CONTROL BLDG SHALL BE CLASSIFIED AS FOLLOWS (SEE CONT 216 SPEC. SECTIONS 15D & 15E.
    A. SEISMIC CATEGORY:
    PIPING & PIPING SUPPORTS = I
    VALVES & ASSOC. COMPONENTS = II
    B. QUALITY CLASS:
    PIPING SUPPORTS = I
    PIPING VALVES & ASSOCIATED COMPONENTS = Q
    PLUMBING FIXTURES HAVE NO SEISMIC REQUIREMENTS.

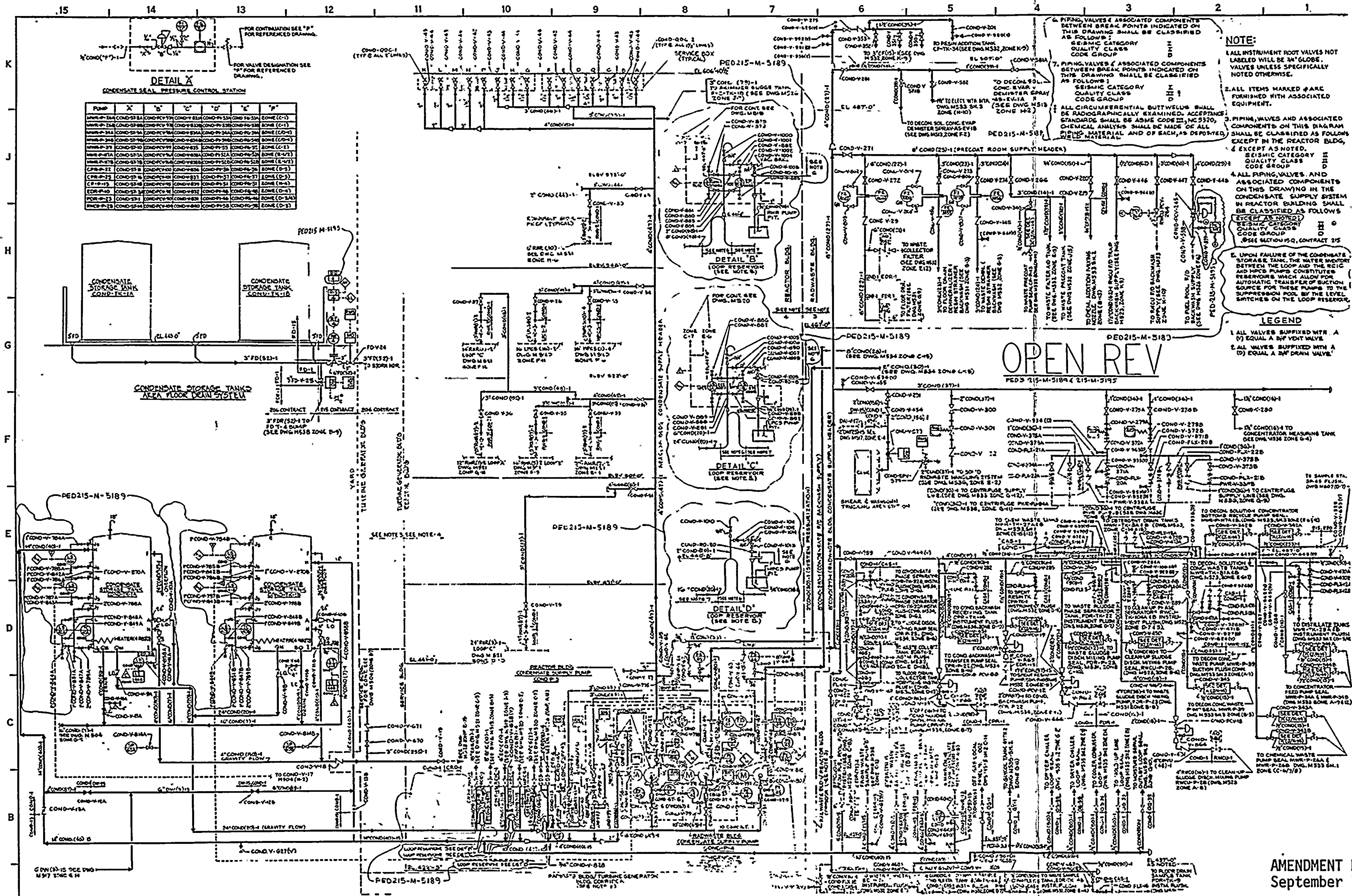
    CODE GROUP FOR ALL PIPING ON THIS DRWG — D
13. ELECTRICAL CONTROLS FOR ELECTRIC HOT WATER HEATER IN ACCORDANCE WITH SPECS.
14. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED & INSTALLED BY OWNER.

LEGEND

H.S. = HOODED SINK
L.S. = LAB. SINK
S.S. = SERVICE SINK
W.C. = WATER CLOSET
LAV. = LAVATORY
SH. = SHOWER
E.W.C. = ELECT. WATER COOLER
J.C. = JANITORS CLOSET
UR. = URINAL
BWV = BACK WATER VALVE

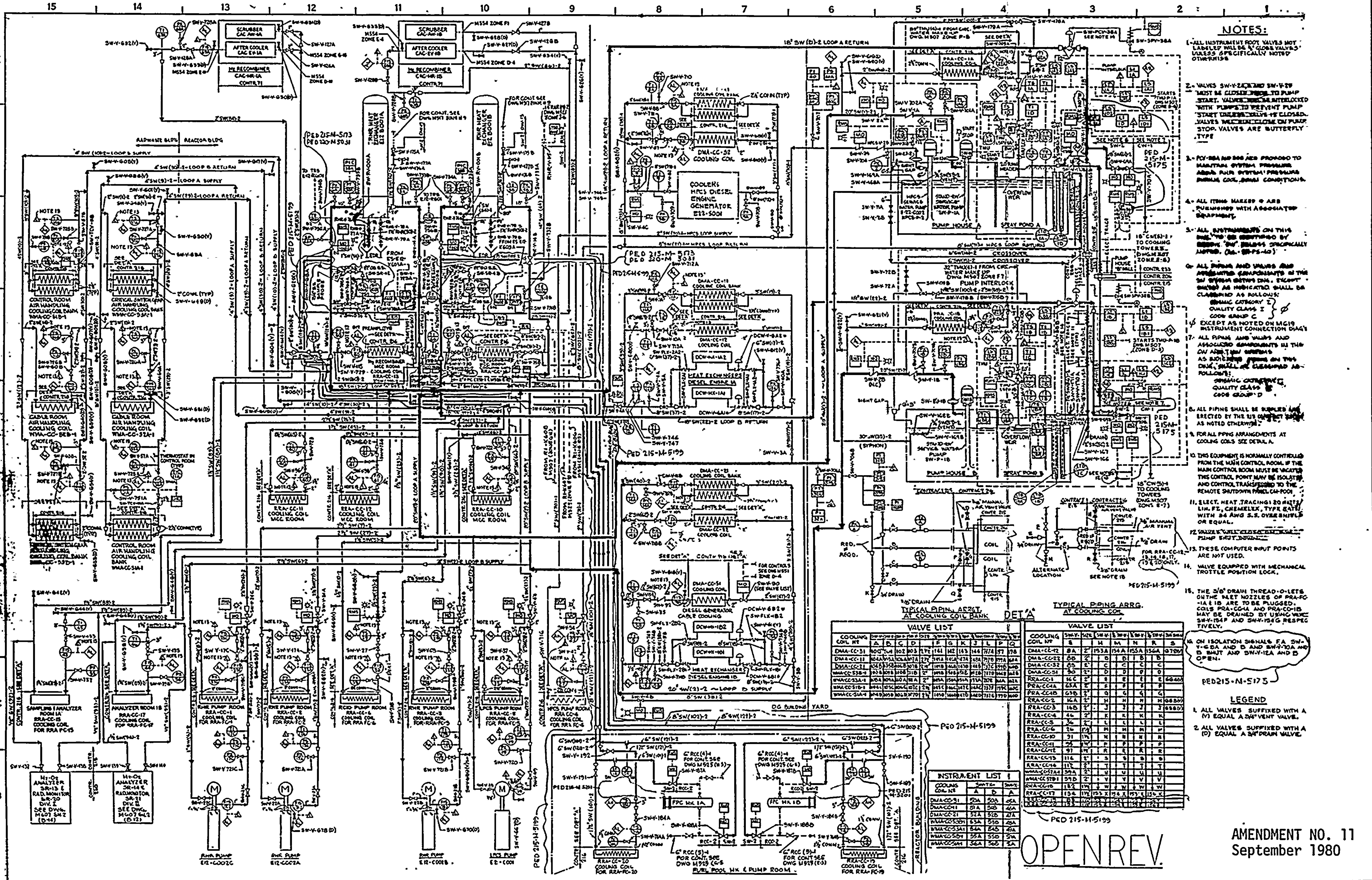
Amendment No. 7
November 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	POTABLE WATER SYSTEM	FIGURE 9.2-5



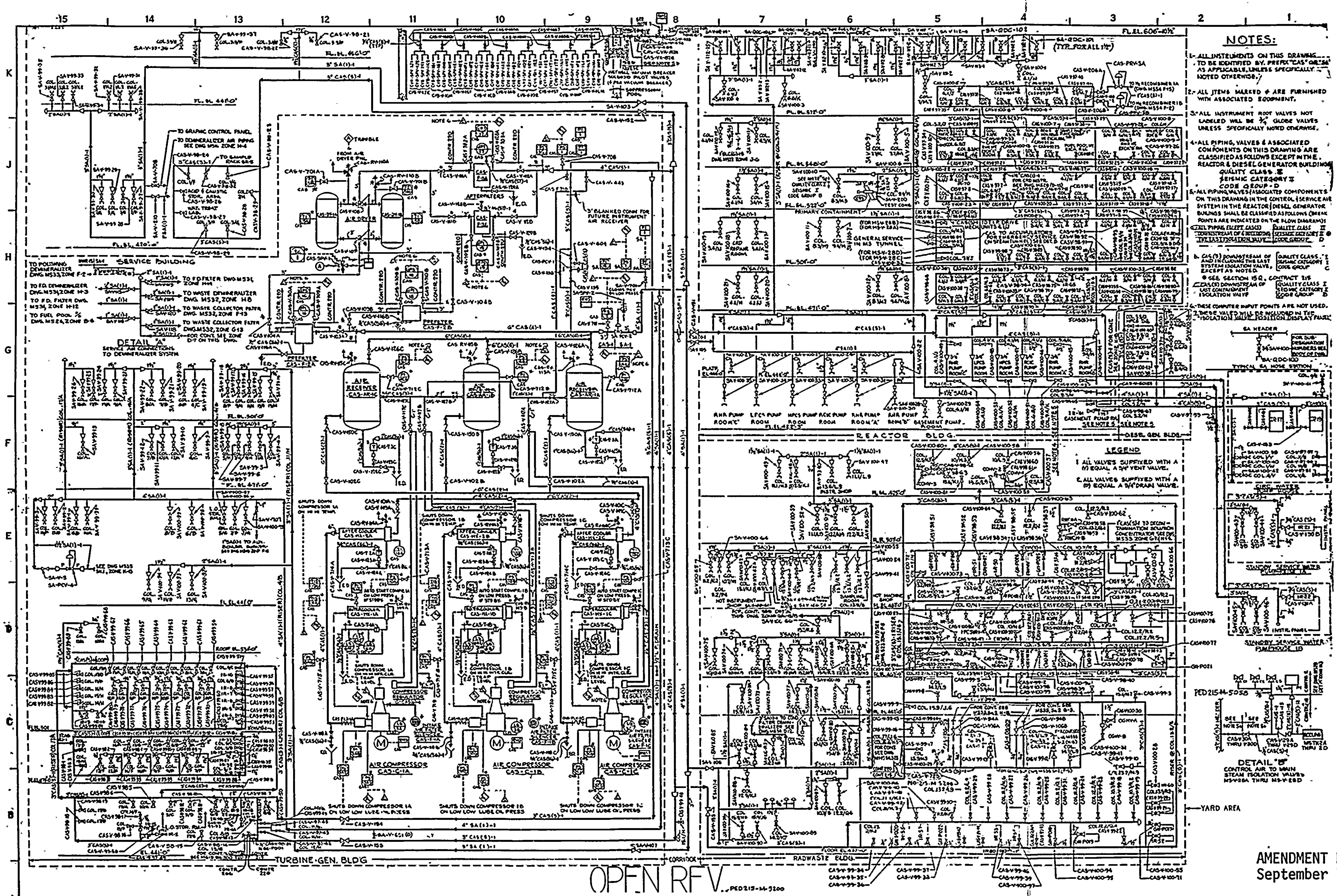
AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONDENSATE SUPPLY SYSTEM	FIGURE 9.2-9



AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY SERVICE WATER SYSTEM	FIGURE 9.2-10

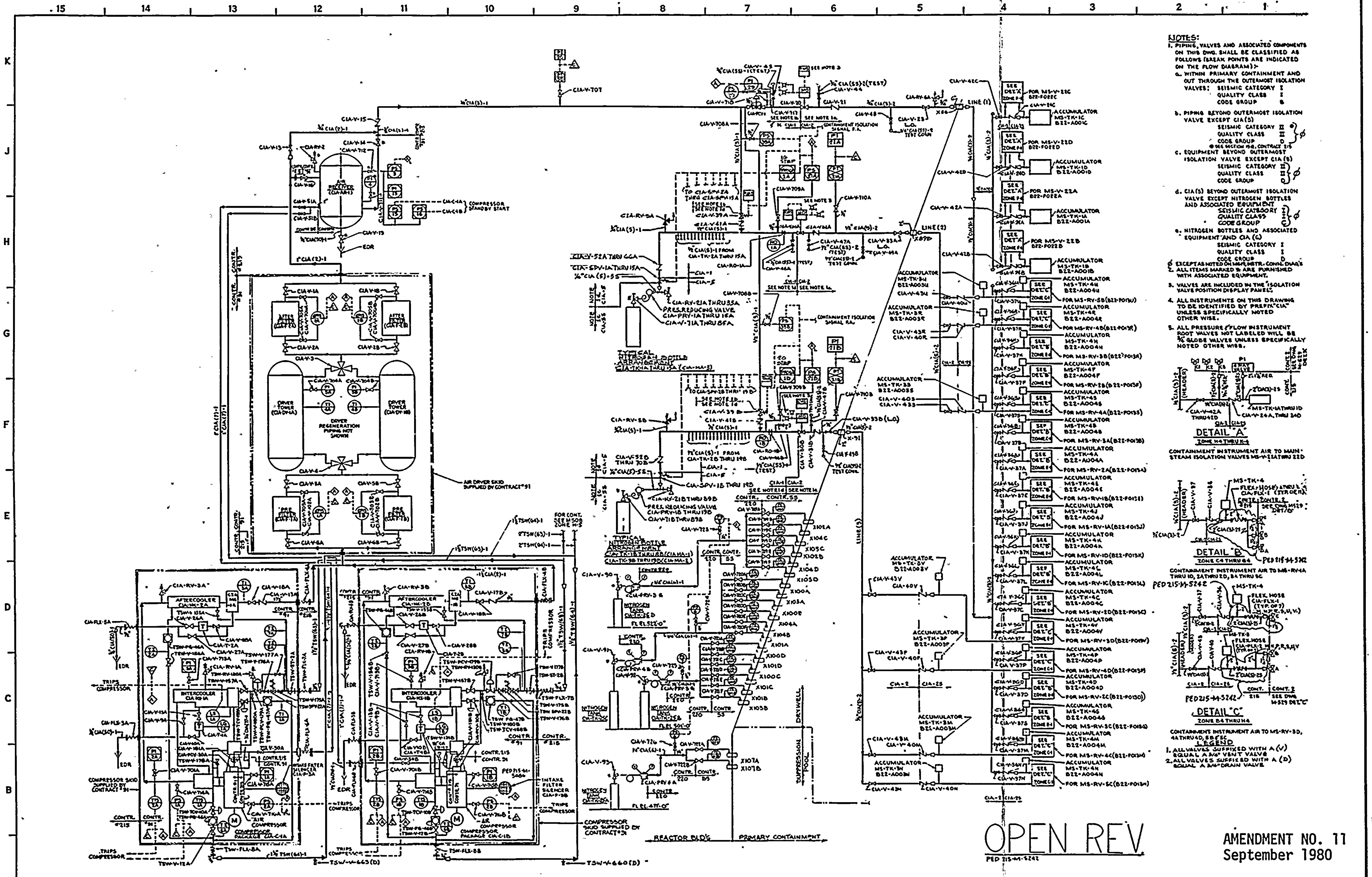


AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTROL AND SERVICE AIR SYSTEM

FIGURE 9.3-1



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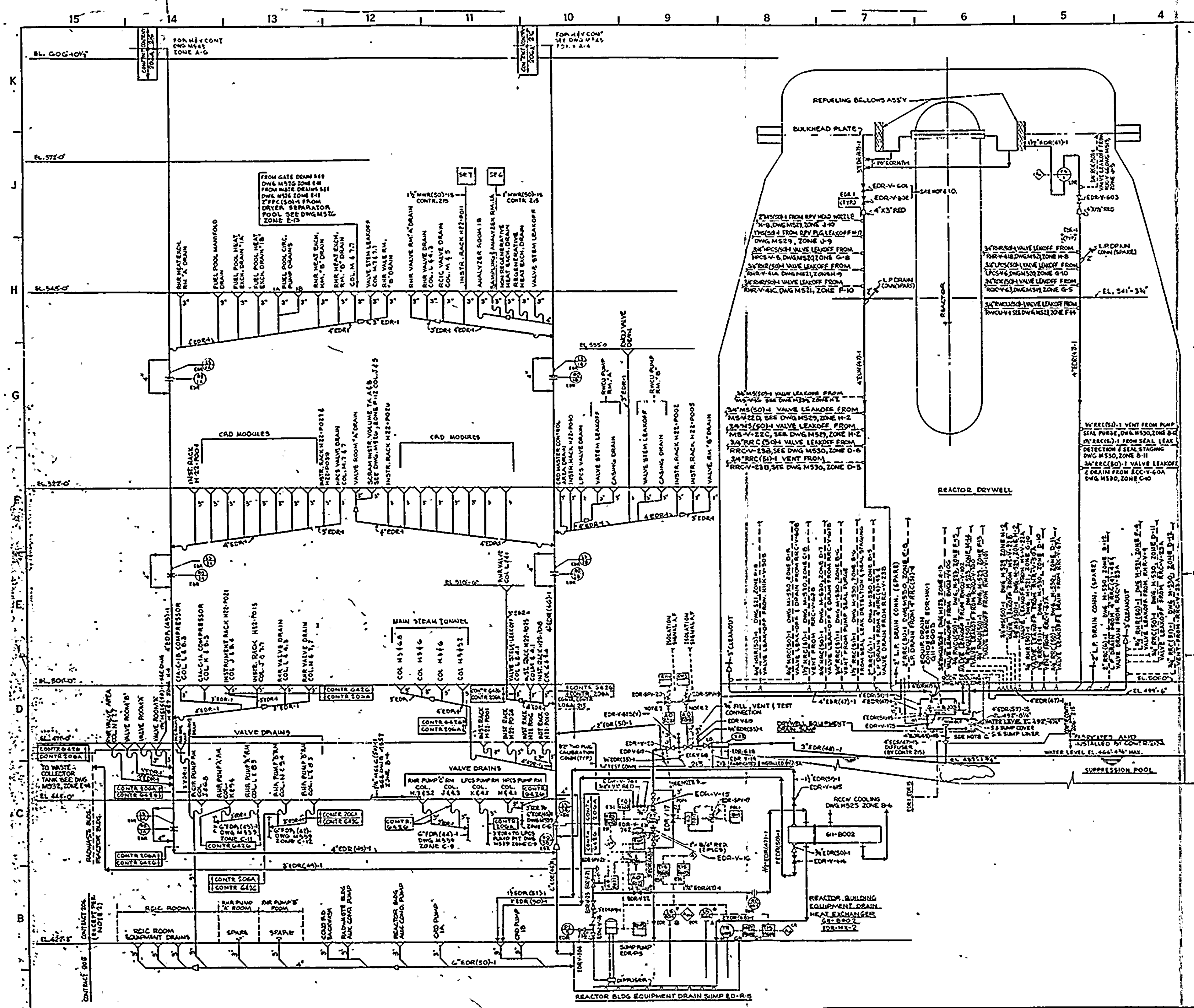
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CONTAINMENT INSTRUMENT AIR SYSTEM	FIGURE 9.3-2



NOTES:

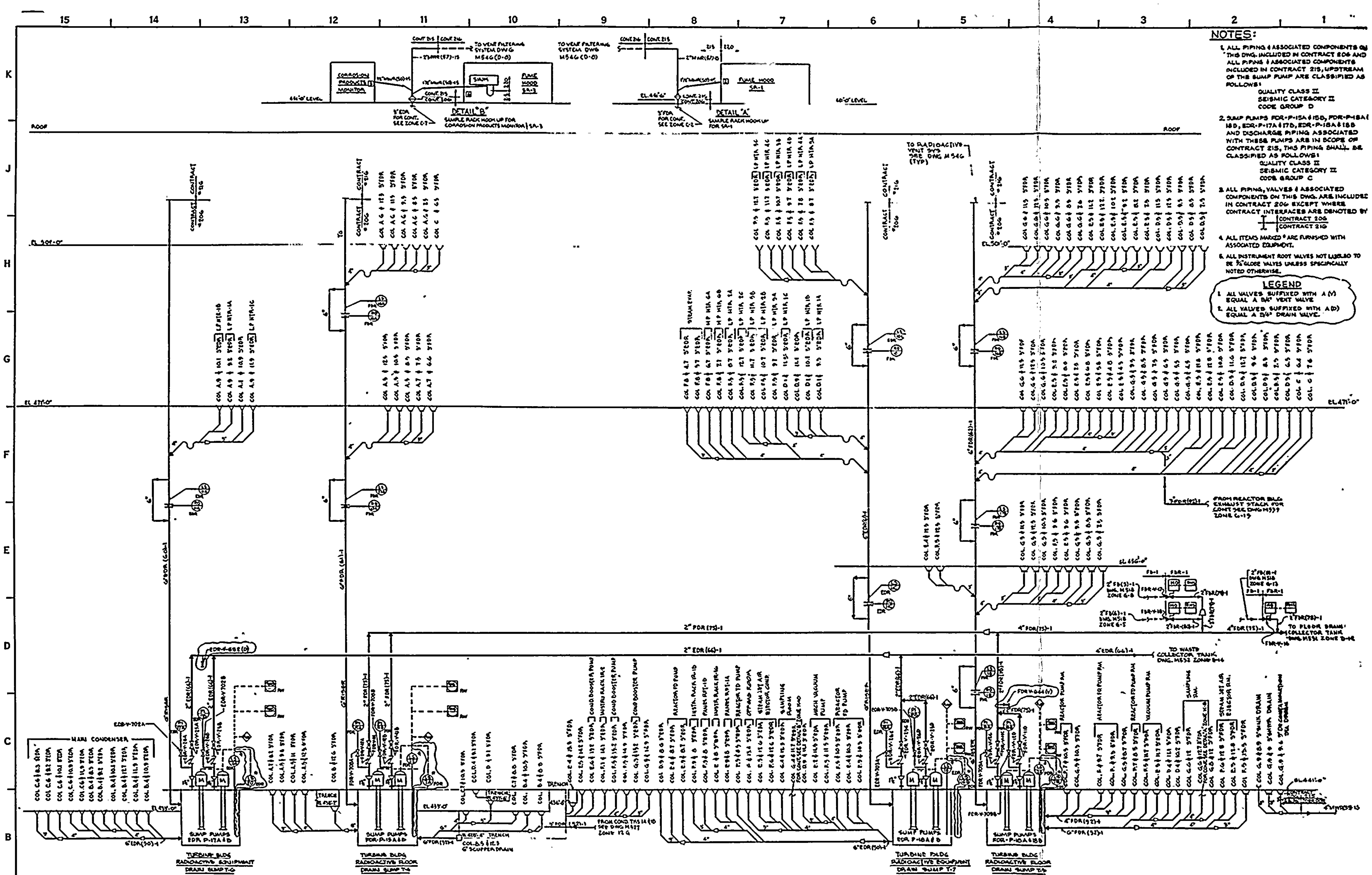
1. EXCEPT AS NOTED IN NOTE 6, ALL PIPING, VALVES & ASSOCIATED COMPONENTS ON THIS DRAWING INCLUDED IN CONTRACTS 205, 206 & 215 ARE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II *
   CODE GROUP D
   * SEE SECTION 15G, CONTRACT 215
2. SUMP PUMP EDR-P-5 AND DISCHARGE PIPING FROM PUMP; REACTOR BUILDING EQUIPMENT DRAIN HEAT EXCHANGER, CONDENSER AND ASSOCIATED PIPING ARE IN SCOPE OF CONTRACT 215.
3. ALL PIPING INSIDE PRIMARY CONTAINMENT EXCEPT AS NOTED, IS IN CONTRACT 215.
4. ALL PIPING & ASSOCIATED COMPONENTS ON THIS DRAWING ARE INCLUDED IN CONTRACT 206 EXCEPT WHERE CONTRACT INTERFACES ARE DENOTED BY:
   CONTRACT 206 (6426)
   CONTRACT 205, 215 OR 216, 206A & 215A
5. (DELETED)
6. PIPING AND ASSOCIATED COMPONENTS IN SUBSYSTEM EDR (48) UPSTREAM OF VALVE EDR-V-20 ARE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS I
   SEISMIC CATEGORY I
   CODE GROUP B
7. THESE VALVES WILL BE INCLUDED IN THE "ISOLATION VALVE POSITION DISPLAY PANEL"
8. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
9. ALL WORK ON THIS DRAWING BY CONTRACT 206A IS SO NOTED. PERMANENT MATERIAL WILL BE FURNISHED TO CONTRACTOR, SEE SPECIFICATIONS.
10. VALVES EDR-V-601 & 602 ARE CLOSED AND TO BE OPENED ONLY WHEN ACCOMMODATING REFUELING OPERATIONS.
11. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.

LEGEND:

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A 3/4" DRAIN VALVE.

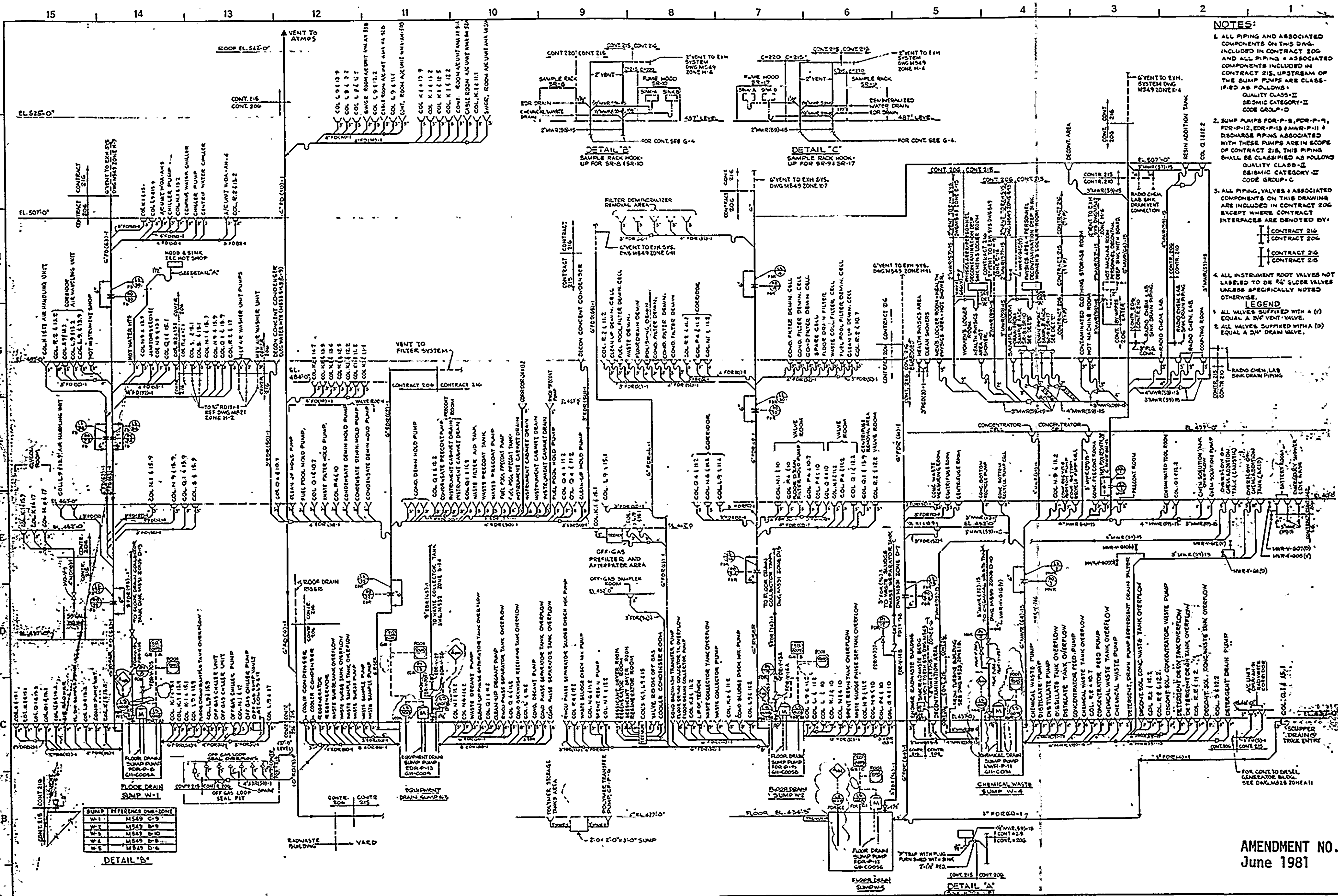
AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	EQUIPMENT DRAIN SYSTEM REACTOR BUILDING	FIGURE 9.3-5



AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TURBINE BUILDING EQUIPMENT AND FLOOR DRAINS	FIGURE 9.3-6



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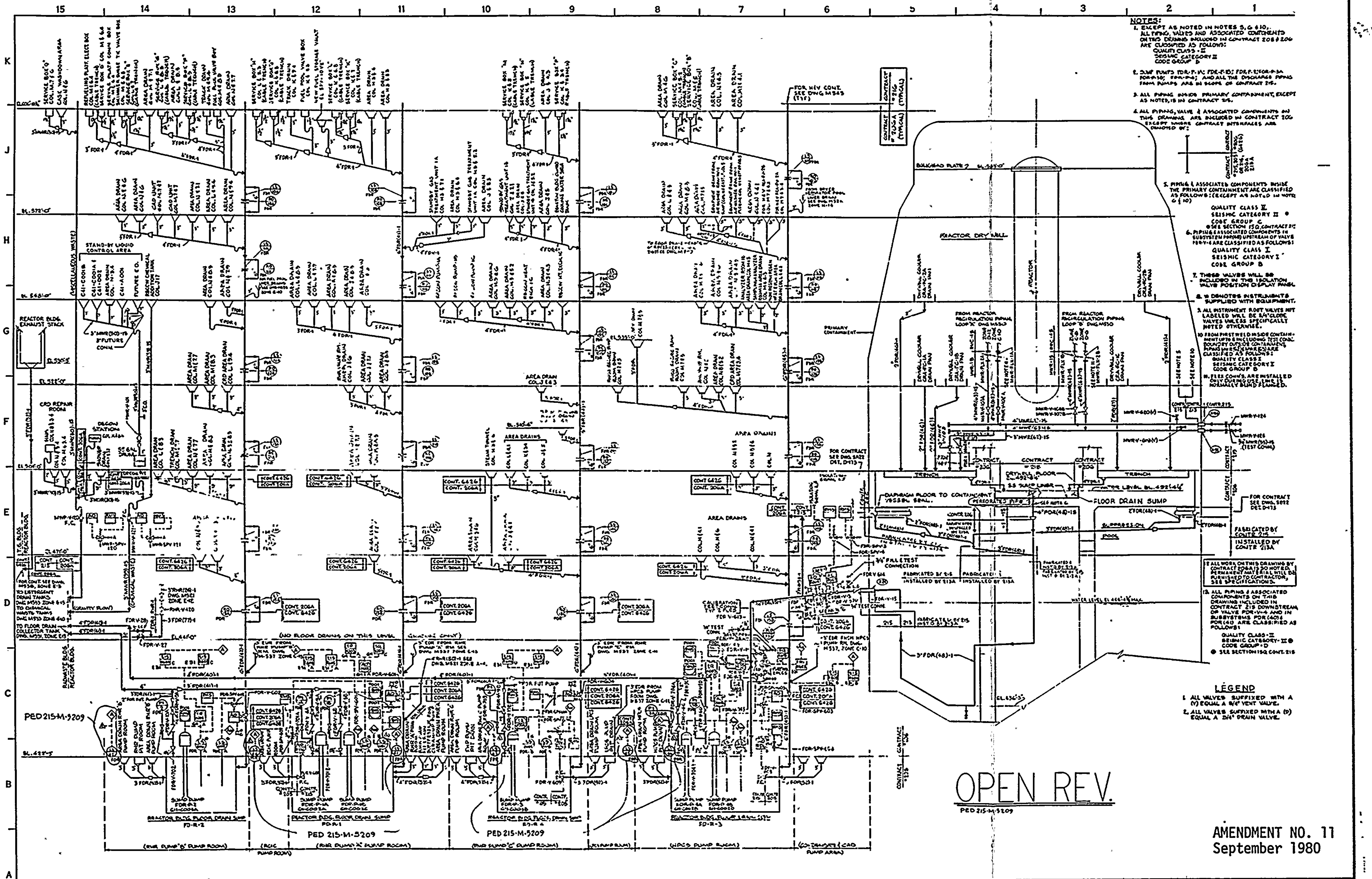
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RADWASTE BUILDING EQUIPMENT AND FLOOR DRAINS	FIGURE 9.3-7



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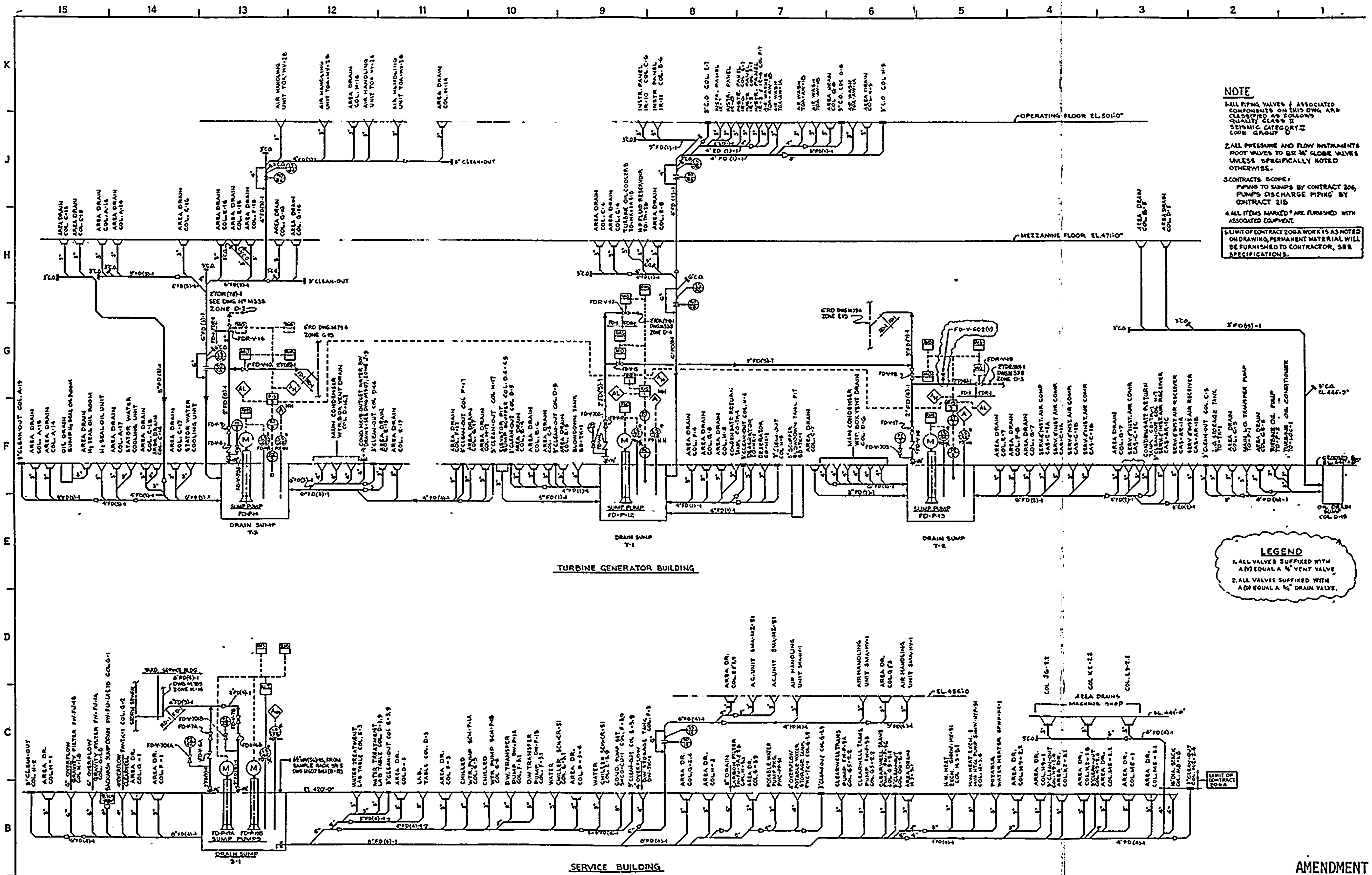
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOOR DRAINS - REACTOR BLDG.	FIGURE 9.3-8



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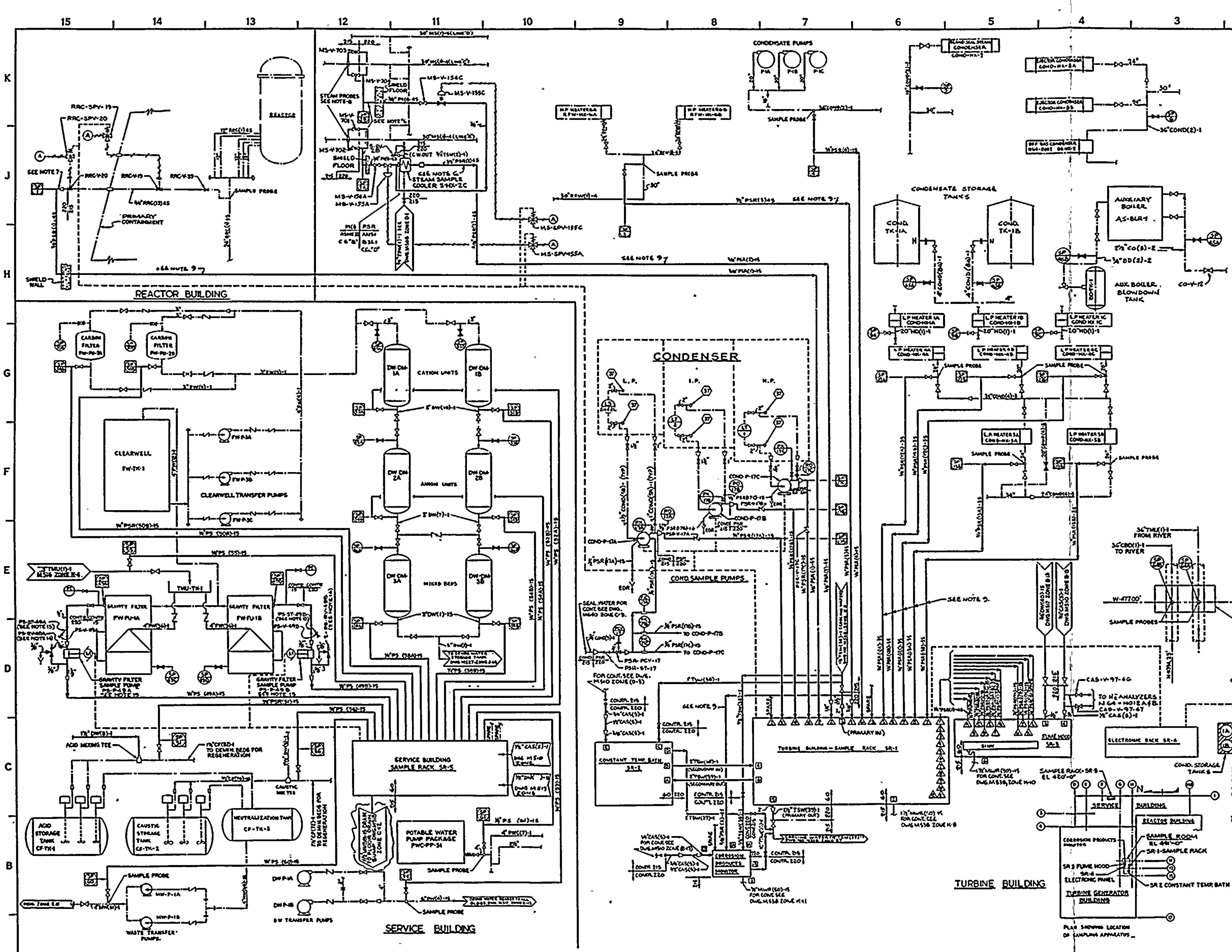
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NON-RADIOACTIVE FLOOR DRAINS	FIGURE 9.3-9



GENERAL NOTES

1. ALL PROCESS PIPING (EXCLUSIVE OF PS & PSR LINE DESIGNATION) IS BY CONTRACT 2808-215. ALL TUBING (PS/PSR DESIGNATION) IS BY CONTRACT 2808-220.
2. SEE DWG. M881 "FLUID SAMPLING LIST" FOR INFORMATION ON SAMPLE POINTS AND INTERNAL CONNECTIONS OF SAMPLE RACKS AND FUME HOODS.
3. ALL SAMPLE LINES TO BE 3/8" O.D. X 0.035" WALL S.S. TUBING, EXCEPT AS NOTED.
4. SAMPLE PROBES SHALL BE INSTALLED IN 4" AND ABOVE PIPING ONLY. FOR PIPING BELOW 4", A PRESSURE TAP SHALL BE USED.
   SAMPLE PROBE SYMBOL
   4" & UP
5. TYPICAL CONTRACT MATCH LINE FOR SAMPLE CONN'S SHALL BE AS FOLLOWS:
   3/4" GLOBE VALVE
   OR
   PROCESS LINE
   REDUCE AS REQD
   CONTRACT 220
   CONTRACT 215
6. ROUTE STEAM SAMPLE LINE FROM LINE TAP, FOR AT LEAST 14 FEET BEHIND SHIELD WALLS BEFORE SAMPLE LINE ENTERS AN ACCESSIBLE AREA.
7. ROUTE REACTOR WATER SAMPLE LINE FROM LINE TAP, FOR AT LEAST 27 FEET BEHIND SHIELD WALLS BEFORE SAMPLE LINE ENTERS AN ACCESSIBLE AREA.
8. SEE DWG. M610 FOR STEAM PROBE INSTALLATION DETAILS.
9. INSULATE FOR PERSONNEL PROTECTION ONLY.
10. SAMPLE PIPING TO BE QUALITY CLASS II AND SEISMIC CATEGORY CLASS II, EXCEPT WHERE NOTED.
11. □ DENOTES SAMPLE RACK SERVICE NOZZLE CONN.
12. △ DENOTES SEQUENTIAL ORDER OF SAMPLE RACK INLET AND OUTLET SAMPLE CONNECTIONS.
13. STRAINERS PS-ST49 A,B ARE C.S. WYE TYPE, 3/8" NPT ENDS, 100 MESH SS STRAINER 150 lb MINIMUM RATING.
14. RELIEF VALVES PS-RV-49 A,B, CS BODY, 3/8 NPT ENDS CORROSION RESISTANT INTERNALS SOFT SEAT SET AT 50 PSIG.
15. PUMPS PS-P-49 A & B TO BE INSTALLED BY CONTR 215

REFERENCE DRAWINGS

1. M881-SHEET 1 THRU 8, FLUID SAMPLING LIST
2. M882-SHEET 1 THRU 20, SCHEMATIC OF SAMPLE RACK
3. M610-INSTALLATION OF SAMPLE PROBES
4. M501-FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS
5. GENERAL ELECTRIC CO. DWGS 106 D4586 & 761E765-FEEDWATER CORROSION PROD. MONITOR.

FLOW DIAGRAMS

M607-SHEET 1-STEAM & LIQUID SAMPLING TURBINE & SERVICE BLD'G
M607-SHEET 2 STEAM & LIQUID SAMPLING REACTOR BLD'G
M607-SHEET 3-STEAM & LIQUID SAMPLING RADWASTE BLD'G.

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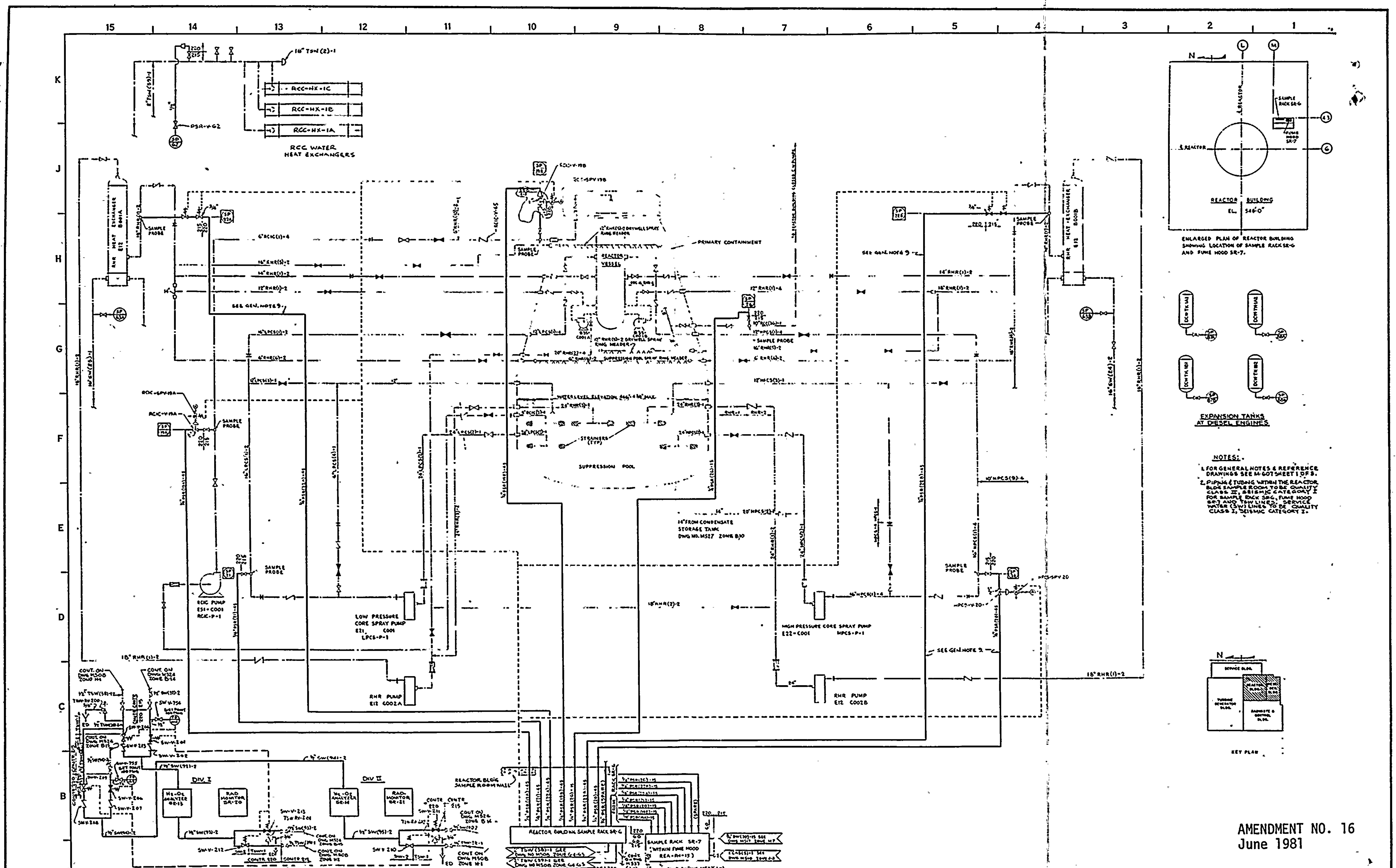
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STEAM AND LIQUID SAMPLING TURBINE AND SERVICE BUILDINGS	FIGURE 9.3-10



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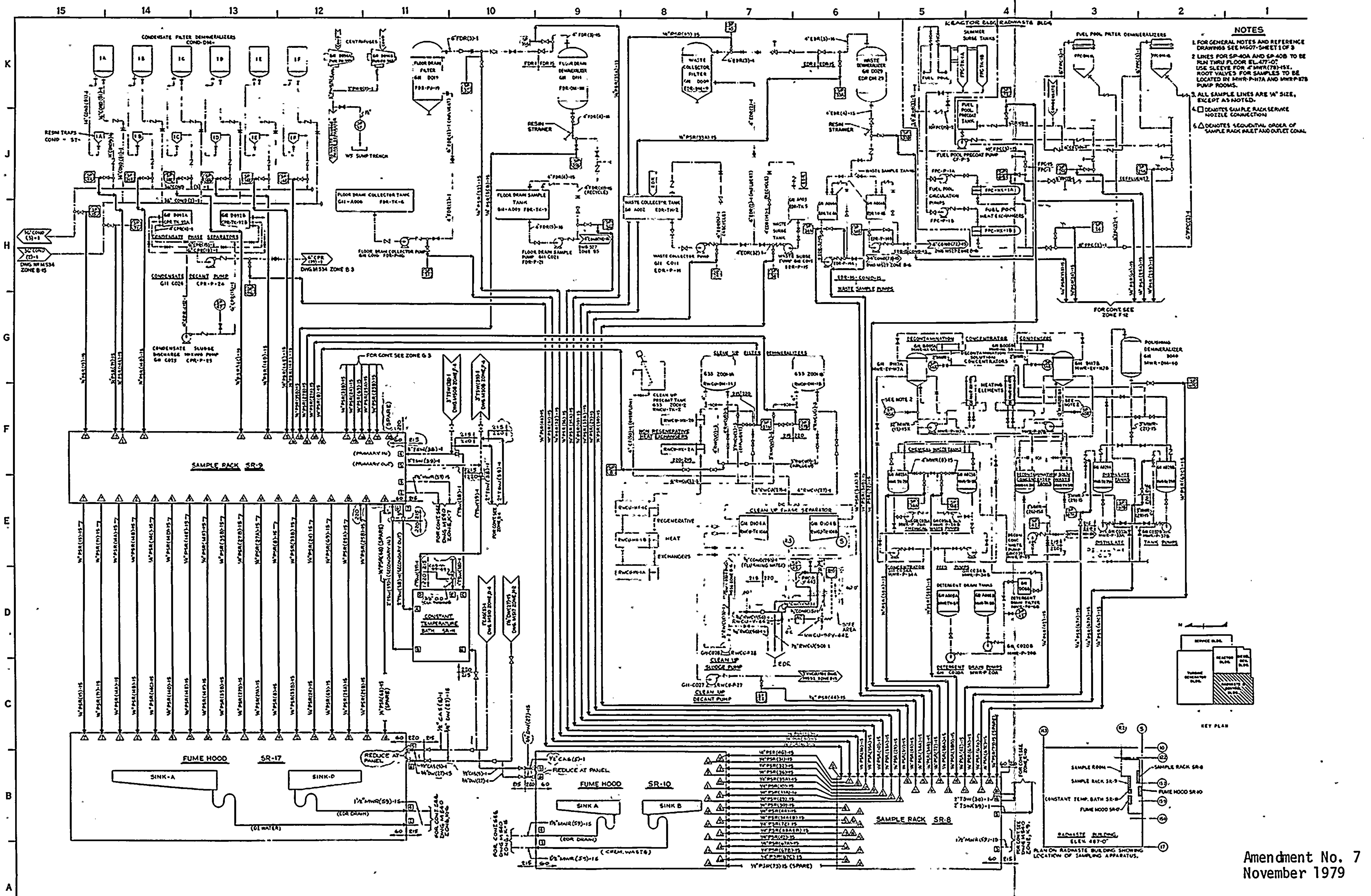
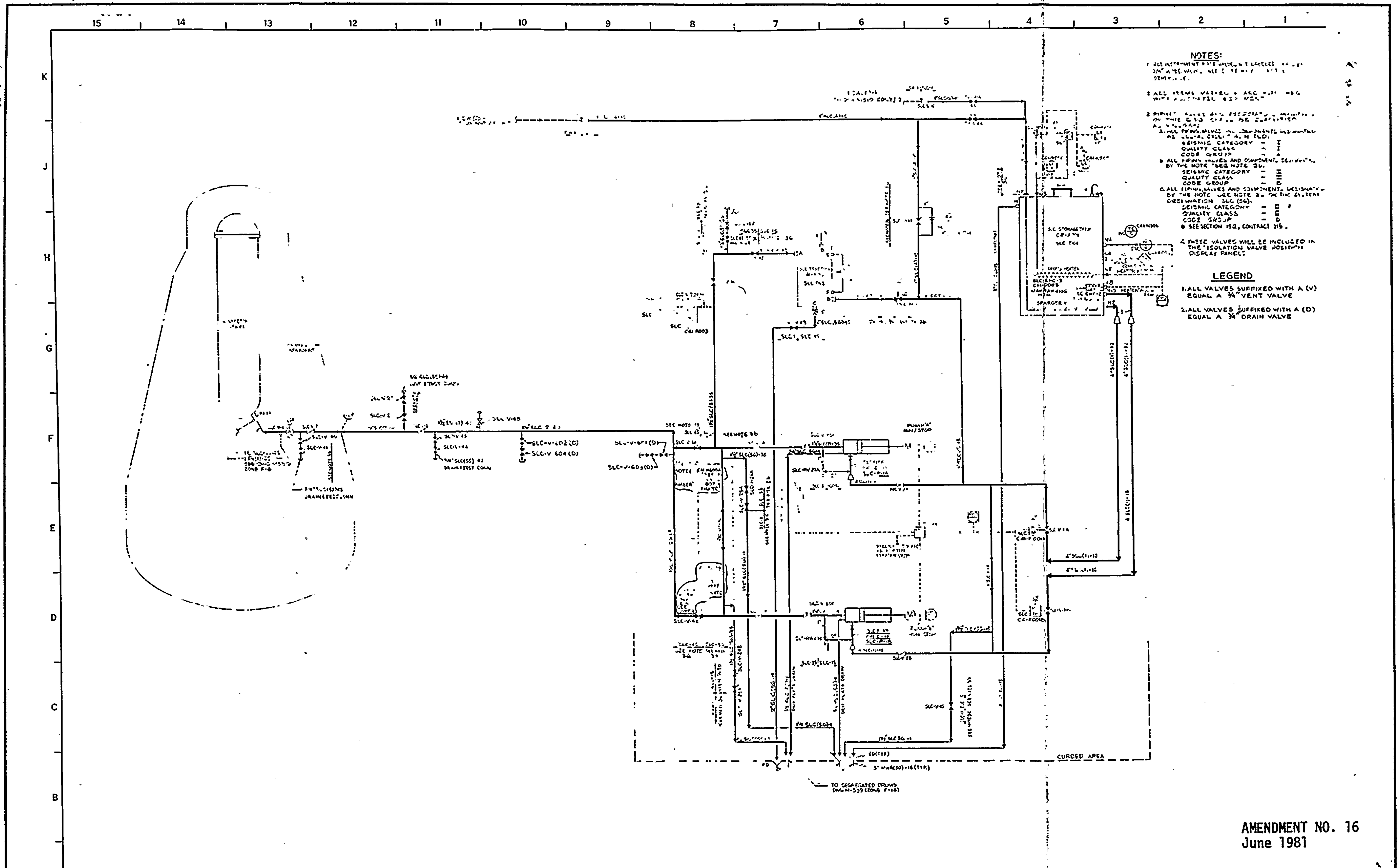
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STEAM AND LIQUID SAMPLING REACTOR BUILDING	FIGURE 9.3-11



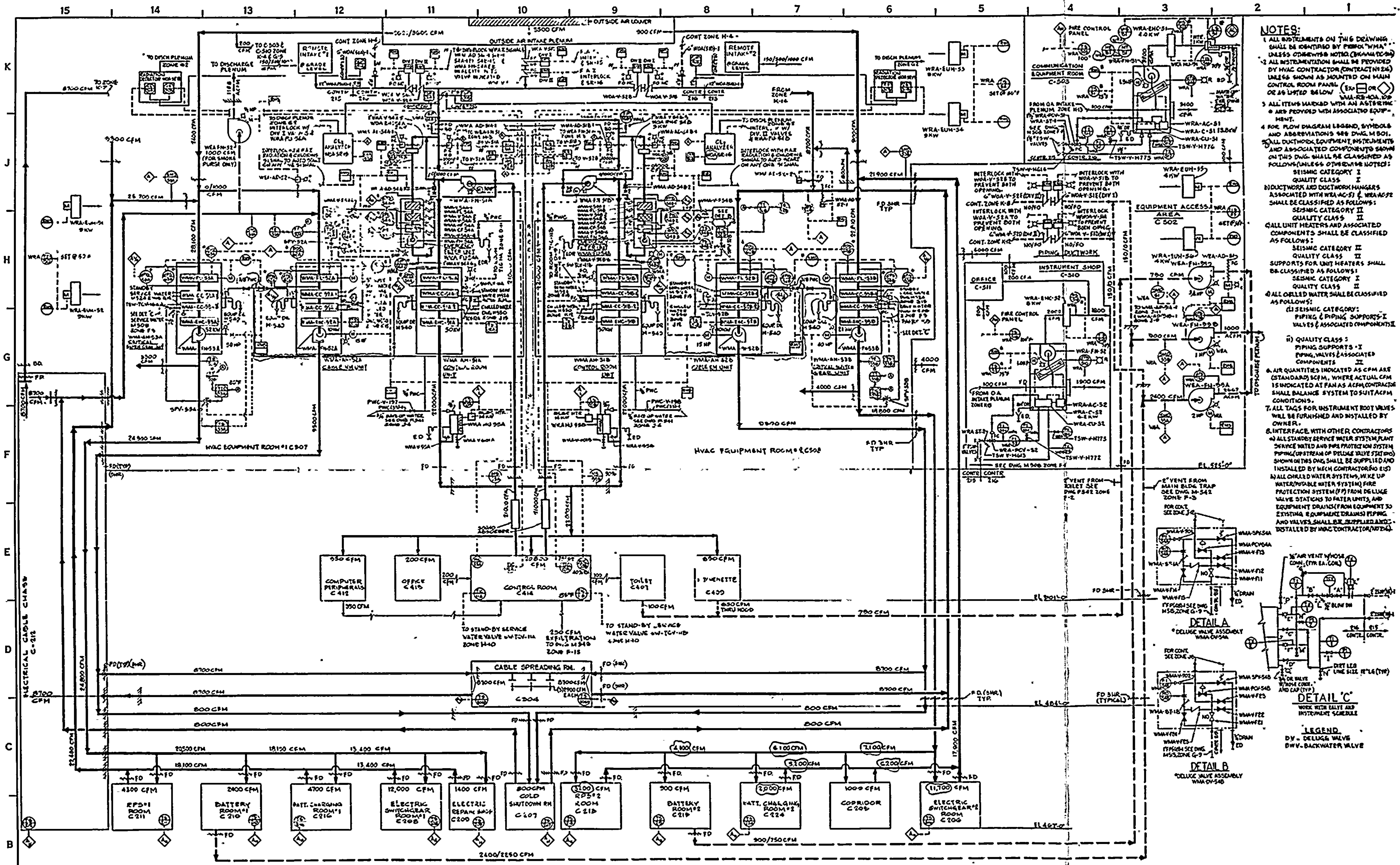
Amendment No. 7
November 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STEAM AND LIQUID SAMPLING RADWASTE BUILDING	FIGURE 9.3-12



AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	STANDBY LIQUID CONTROL SYSTEM REACTOR BLDG.	FIGURE 9.3-13



NOTES:

1. ALL INSTRUMENTS ON THIS DRAWING SHALL BE IDENTIFIED BY PREFIX "WMA" UNLESS OTHERWISE NOTED (EX-WMA-TC-34)
2. ALL INSTRUMENTATION SHALL BE PROVIDED BY HVAC CONTRACTOR (CONTRACT N° 216) UNLESS SHOWN AS MOUNTED ON MAIN CONTROL ROOM PANEL (EX- ⊟ OR ◇) OR AS LISTED BELOW WMA-RE-40A, 40B
3. ALL ITEMS MARKED WITH AN ASTERISK ✱ ARE PROVIDED WITH ASSOCIATED EQUIPMENT.
4. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG. M 501.
5. a) ALL DUCTWORK, EQUIPMENT, INSTRUMENTS AND ASSOCIATED COMPONENTS SHOWN ON THIS DWG. SHALL BE CLASSIFIED AS FOLLOWS (UNLESS OTHERWISE NOTED):
   SEISMIC CATEGORY I
   QUALITY CLASS I
   b) DUCTWORK AND DUCTWORK HANGERS ASSOCIATED WITH WRA-AC-51 & WRA-AC-52 SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY II
   QUALITY CLASS II
   c) ALL UNIT HEATERS AND ASSOCIATED COMPONENTS SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY II
   QUALITY CLASS II
   SUPPORTS FOR UNIT HEATERS SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY I
   QUALITY CLASS II
   d) ALL CHILLED WATER SHALL BE CLASSIFIED AS FOLLOWS:
   i) SEISMIC CATEGORY:
   PIPING & PIPING SUPPORTS-I
   VALVES & ASSOCIATED COMPONENTS-II
   ii) QUALITY CLASS:
   PIPING SUPPORTS -I
   PIPING, VALVES & ASSOCIATED COMPONENTS II
6. AIR QUANTITIES INDICATED AS CFM ARE (STANDARD) SCFM. WHERE ACTUAL CFM IS INDICATED AT FAN AS ACFM, CONTRACTOR SHALL BALANCE SYSTEM TO SUIT ACFM CONDITIONS.
7. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED AND INSTALLED BY OWNER.
8. INTERFACE WITH OTHER CONTRACTORS
   a) ALL STANDBY SERVICE WATER SYSTEM, PLANT SERVICE WATER AND FIRE PROTECTION SYSTEM PIPING (UPSTREAM OF DELUGE VALVE STATIONS) SHOWN ON THIS DWG. SHALL BE SUPPLIED AND INSTALLED BY MECH CONTRACTOR (NO 215)
   b) ALL CHILLED WATER SYSTEMS, MAKE UP WATER (POTABLE WATER SYSTEM), FIRE PROTECTION SYSTEM (FP) FROM DELUGE VALVE STATIONS TO FILTER UNITS, AND EQUIPMENT DRAINS (FROM EQUIPMENT TO EXISTING EQUIPMENT DRAINS) PIPING AND VALVES SHALL BE SUPPLIED AND INSTALLED BY HVAC CONTRACTOR (NO 216).

CRITICAL SWITCHGEAR AIR HANDLING UNITS - VALVE AND INSTRUMENT SCHEDULE

DRAIN VALVE "Q"	VENT VALVE "P"	DRAIN VALVE "N"	DRAIN VALVE "M"	BLOWDOWN VALVE "L"	ROOT VALVE "K"	ROOT VALVE "J"	UNIT NO.	TCV	SHUTOFF VALVES "A"-"B"-"C"-"D"	BALANCING VALVES "E"-"F"	BYPASS VALVE "G"	TI 1-2	TX 1-2	PI 1-2	FX 1	ROOT VALVE "H"	ROOT VALVE "I"
TSW-V-H615A	TSW-V-H779A	TSW-V-H612A	TSW-V-H611A	TSW-V-H610A	TSW-V-H762A	TSW-V-H761A	WMA-AH-53A	TSW-TCV-16A	2" TSW-H53A-A,B,C,D	2" TSW-H53A-E,F	1½" TSW-H53A-G	TSW 53A-1,2	TSW-53A-1,2	TSW-53A-1,2	TSW 53A-1	TSW-V-H753A	TSW-V-H754A
TSW-V-H615B	TSW-V-H779B	TSW-V-H612B	TSW-V-H611B	TSW-V-H610B	TSW-V-H762B	TSW-V-H761B	WMA-AH-53B	TSW-TCV-16B	2" TSW-H53B-A,B,C,D	2" TSW-H53B-E,F	1½" TSW-H53B-G	TSW 53B-1,2	TSW 53B-1,2	TSW 53B-1,2	TSW 53B-1	TSW-V-H753B	TSW-V-H754B

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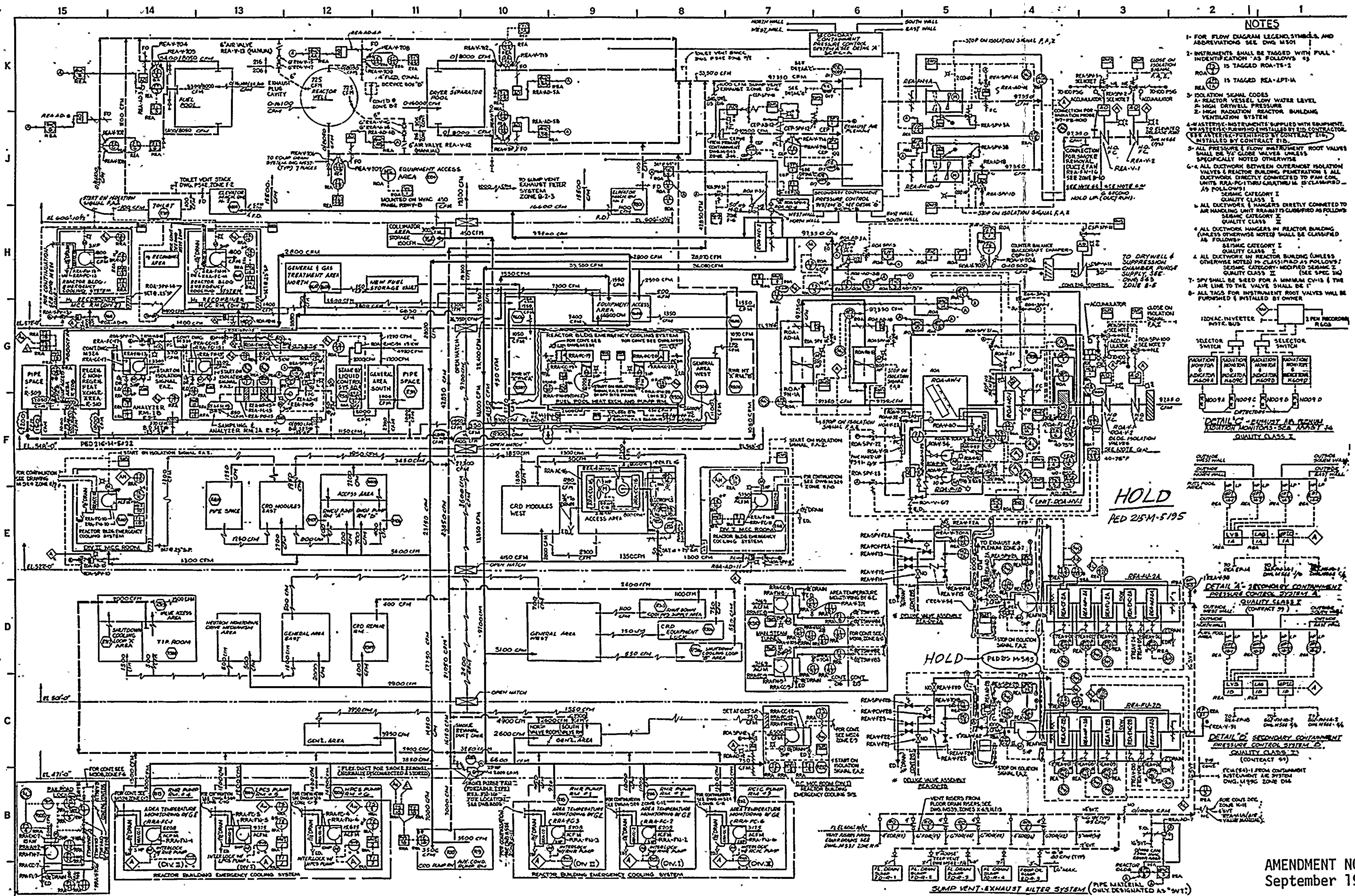
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

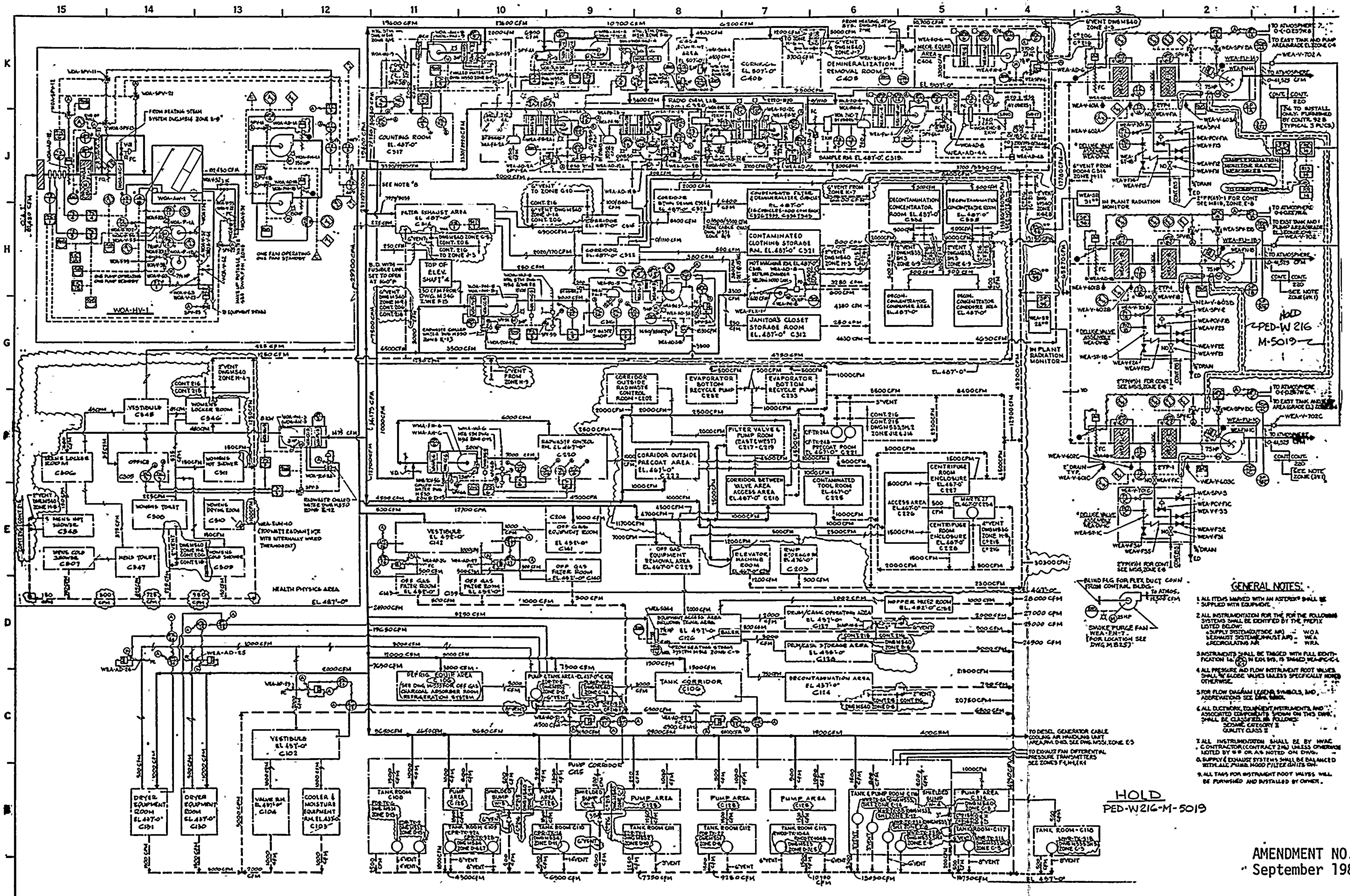
HVAC CONTROL ROOM & CRITICAL SWITCHGEAR

FIGURE 9.4-1



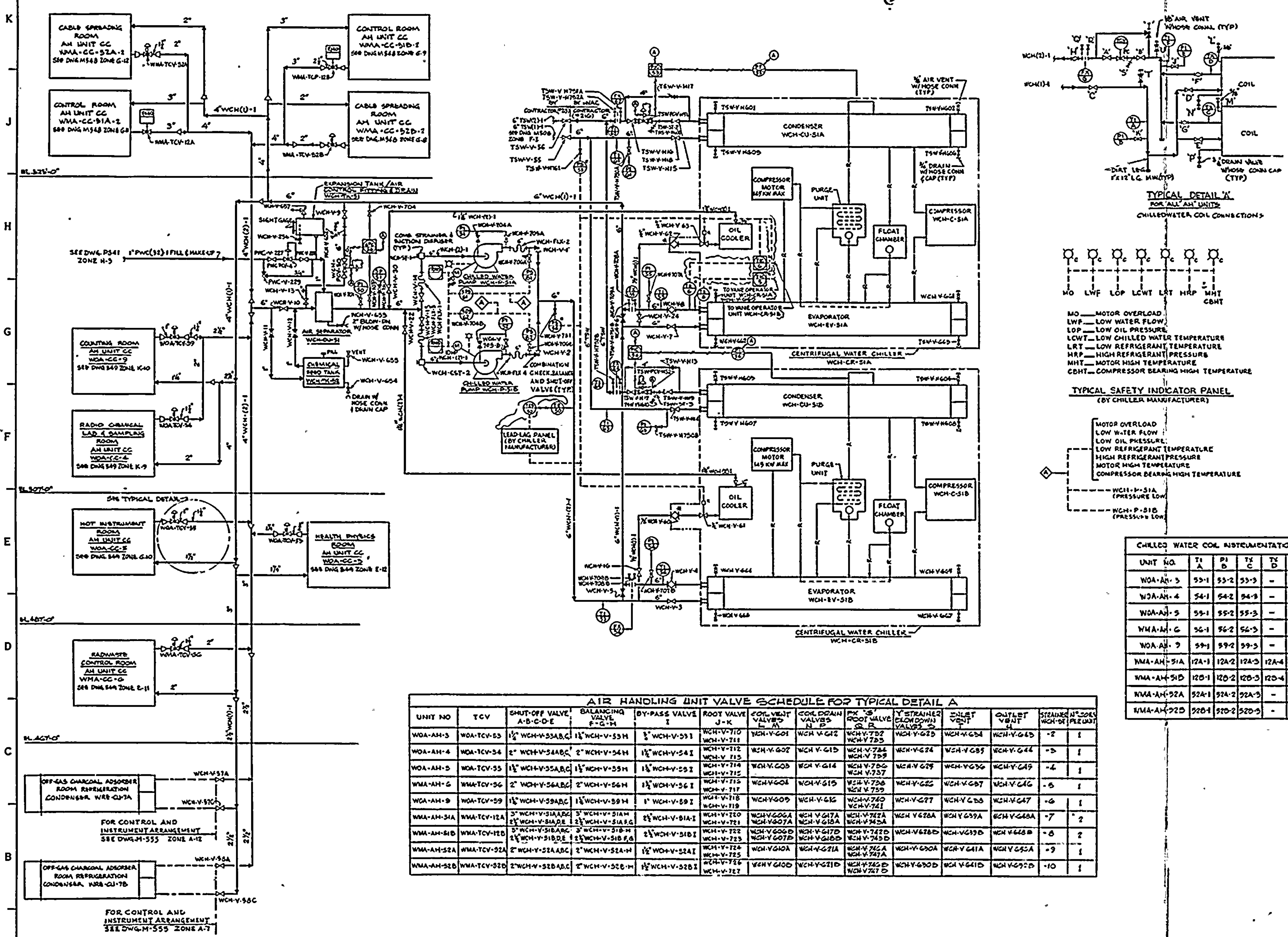
AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC SYSTEMS FOR REACTOR BUILDING	FIGURE 9.4-2



AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC SYSTEMS FOR RADWASTE BUILDING	FIGURE 9.4-3



GENERAL NOTES

1. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG.M501.
2. ALL INSTRUMENTATION AND CONTROLS SHALL BE BY HVAC CONTRACTOR (CONTRACT Nº 216)
3. ALL INSTRUMENTATION AND CONTROLS FOR CHILLED WATER SYSTEM SHALL BE IDENTIFIED BY PREFIX "WCH" UNLESS OTHERWISE NOTED
4. ALL ITEMS MARKED WITH AN ASTERISK * ARE AVAILABLE WITH ASSOCIATED EQUIPMENT.
5. ALL PIPING, PIPING SUPPORTS, VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS

   SEISMIC CATEGORY:

   PIPING AND PIPING SUPPORTS IN CONTROL BLDG. — I

   PIPING AND PIPING SUPPORTS OUTSIDE CONTROL BLDG. — II

   VALVES AND ASSOCIATED COMPONENTS - II

   QUALITY CLASS:

   PIPING SUPPORTS IN CONTROL BLDG. — I

   PIPING SUPPORTS OUTSIDE CONTROL BLDG. — II

   PIPING, VALVES AND ASSOCIATED COMPONENTS — II
6. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED AND INSTALLED BY OWNER
7. LINES SHOWN (— R —) ARE INTERCONNECTING REFRIGERANT LINES PROVIDED BY THE CHILLER MANUFACTURER.
8. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE 3/4" GLOBE VALVES UNLESS OTHERWISE NOTED

CHILLED WATER COIL INSTRUMENTATION NUMBERS

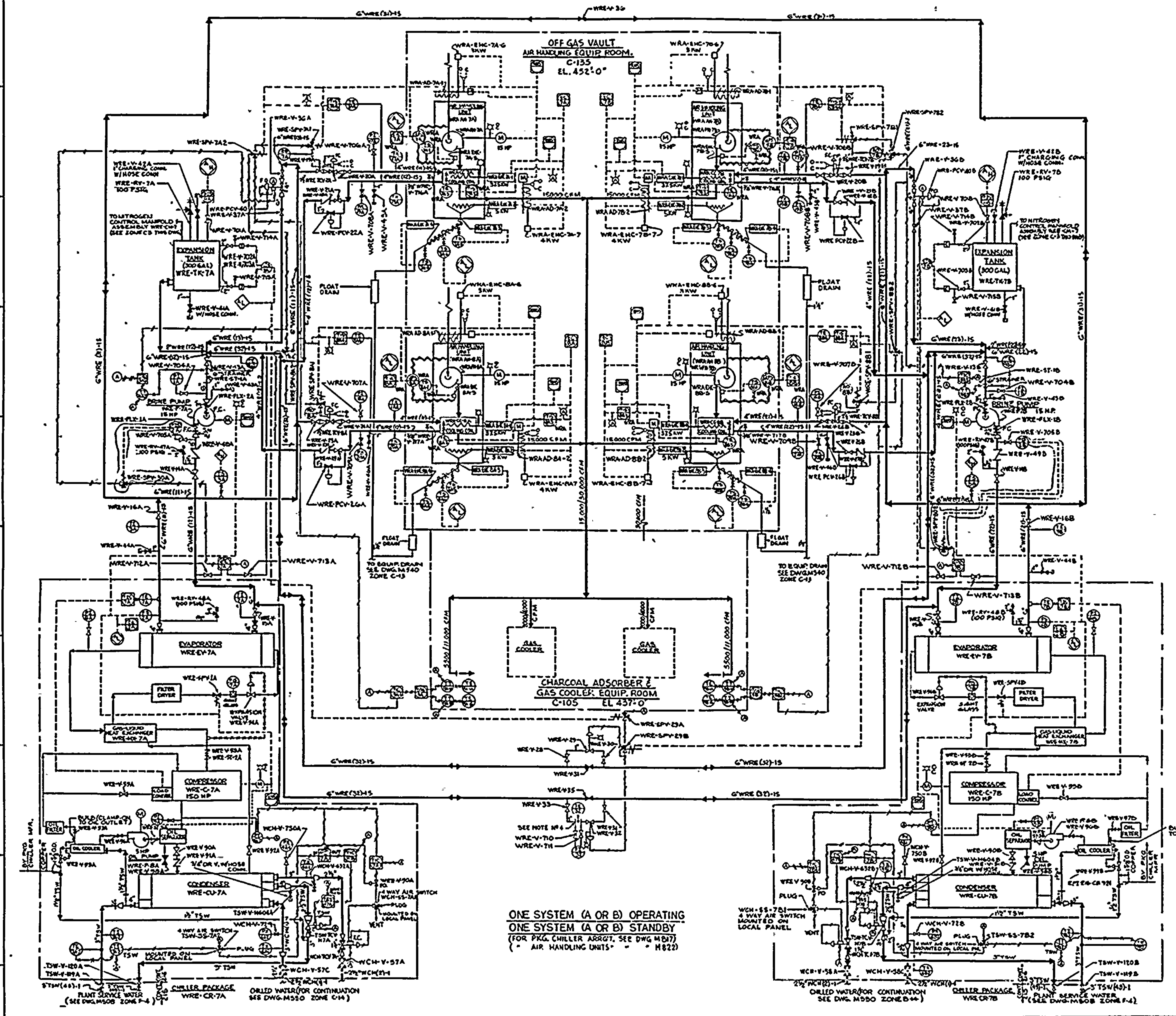
UNIT NO.	TI A	PI B	TX C	TX D	PI E	TI F	FX G
WOA-AH-3	53-1	53-2	53-3	-	53-5	53-6	53-7
WOA-AH-4	54-1	54-2	54-3	-	54-5	54-6	54-7
WOA-AH-5	55-1	55-2	55-3	-	55-5	55-6	55-7
WMA-AH-6	56-1	56-2	56-3	-	56-5	56-6	56-7
WOA-AH-9	59-1	59-2	59-3	-	59-5	59-6	59-7
WMA-AH-51A	12A-1	12A-2	12A-3	12A-4	12A-5	12A-6	12A-7
WMA-AH-51B	12B-1	12B-2	12B-3	12B-4	12B-5	12B-6	12B-7
WMA-AH-52A	52A-1	52A-2	52A-3	-	52A-5	52A-6	52A-7
WMA-AH-52B	52B-1	52B-2	52B-3	-	52B-5	52B-6	52B-7

AIR HANDLING UNIT VALVE SCHEDULE FOR TYPICAL DETAIL A

UNIT NO	TCV	SHUT-OFF VALVE A-B-C-D-E	BALANCING VALVE F-G-H	BY-PASS VALVE I	ROOT VALVE J-K	COIL VENT VALVES L M	COIL DRAIN VALVES N P	FX "G" ROOT VALVE Q R	"Y" STRAINER BLOWDOWN VALVES S	INLET VENT T	OUTLET VENT U	STRAINER WCH-ST	Nº COILS PER UNIT
WOA-AH-3	WOA-TCV-53	1¼" WCH-V-53A,B,C	1¼" WCH-V-53H	¾" WCH-V-53I	WCH-V-710 WCH-V-711	WCH-V-601	WCH-V-612	WCH-V-732 WCH-V-733	WCH-V-623	WCH-V-634	WCH-V-643	-2	1
WOA-AH-4	WOA-TCV-54	2" WCH-V-54A,B,C	2" WCH-V-54H	1½" WCH-V-54I	WCH-V-712 WCH-V-713	WCH-V-602	WCH-V-613	WCH-V-734 WCH-V-735	WCH-V-624	WCH-V-635	WCH-V-644	-3	1
WOA-AH-5	WOA-TCV-55	1½" WCH-V-55A,B,C	1½" WCH-V-55H	1½" WCH-V-55I	WCH-V-714 WCH-V-715	WCH-V-603	WCH-V-614	WCH-V-736 WCH-V-737	WCH-V-625	WCH-V-636	WCH-V-645	-4	1
WMA-AH-6	WMA-TCV-56	2" WCH-V-56A,B,C	2" WCH-V-56H	1½" WCH-V-56I	WCH-V-716 WCH-V-717	WCH-V-604	WCH-V-615	WCH-V-738 WCH-V-739	WCH-V-626	WCH-V-637	WCH-V-646	-5	1
WOA-AH-9	WOA-TCV-59	1½" WCH-V-59A,B,C	1½" WCH-V-59H	1" WCH-V-59I	WCH-V-718 WCH-V-719	WCH-V-605	WCH-V-616	WCH-V-740 WCH-V-741	WCH-V-627	WCH-V-638	WCH-V-647	-6	1
WMA-AH-51A	WMA-TCV-12A	3" WCH-V-51A,A,B,C 2½" WCH-V-51A,D,E	3" WCH-V-51AH 2½" WCH-V-51A F,G	2½" WCH-V-51A-I	WCH-V-720 WCH-V-721	WCH-V-606A WCH-V-607A	WCH-V-617A WCH-V-618A	WCH-V-742A WCH-V-743A	WCH-V-628A	WCH-V-639A	WCH-V-648A	-7	* 2
WMA-AH-51B	WMA-TCV-12B	3" WCH-V-51B,A,B,C 2½" WCH-V-51B,D,E	3" WCH-V-51B-H 2½" WCH-V-51B F,G	2½" WCH-V-51BI	WCH-V-722 WCH-V-723	WCH-V-606B WCH-V-607B	WCH-V-617B WCH-V-618B	WCH-V-742B WCH-V-743B	WCH-V-628B	WCH-V-639B	WCH-V-648B	-8	2
WMA-AH-52A	WMA-TCV-52A	2" WCH-V-52A A,B,C	2" WCH-V-52A-H	1½" WCH-V-52AI	WCH-V-724 WCH-V-725	WCH-V-610A	WCH-V-621A	WCH-V-746A WCH-V-747A	WCH-V-630A	WCH-V-641A	WCH-V-652A	-9	1
WMA-AH-52B	WMA-TCV-52B	2" WCH-V-52B A,B,C	2" WCH-V-52B-H	1½" WCH-V-52BI	WCH-V-726 WCH-V-727	WCH-V-610B	WCH-V-621B	WCH-V-746B WCH-V-747B	WCH-V-630B	WCH-V-641B	WCH-V-652B	-10	1

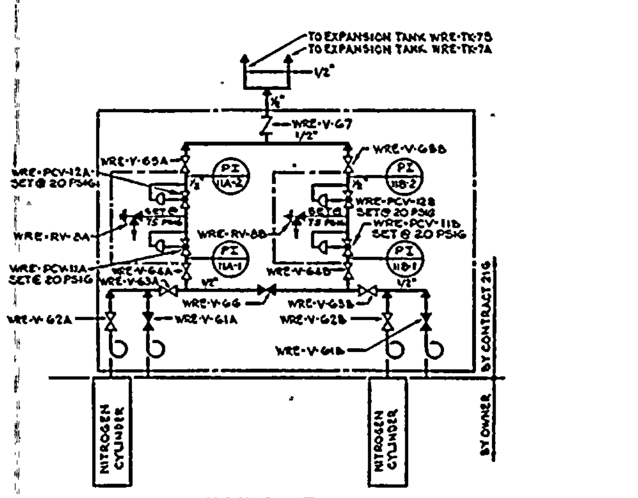
Amendment No. 7
November 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC CHILLED WATER SYSTEM IN RADWASTE BUILDING	FIGURE 9.4-4



NOTES

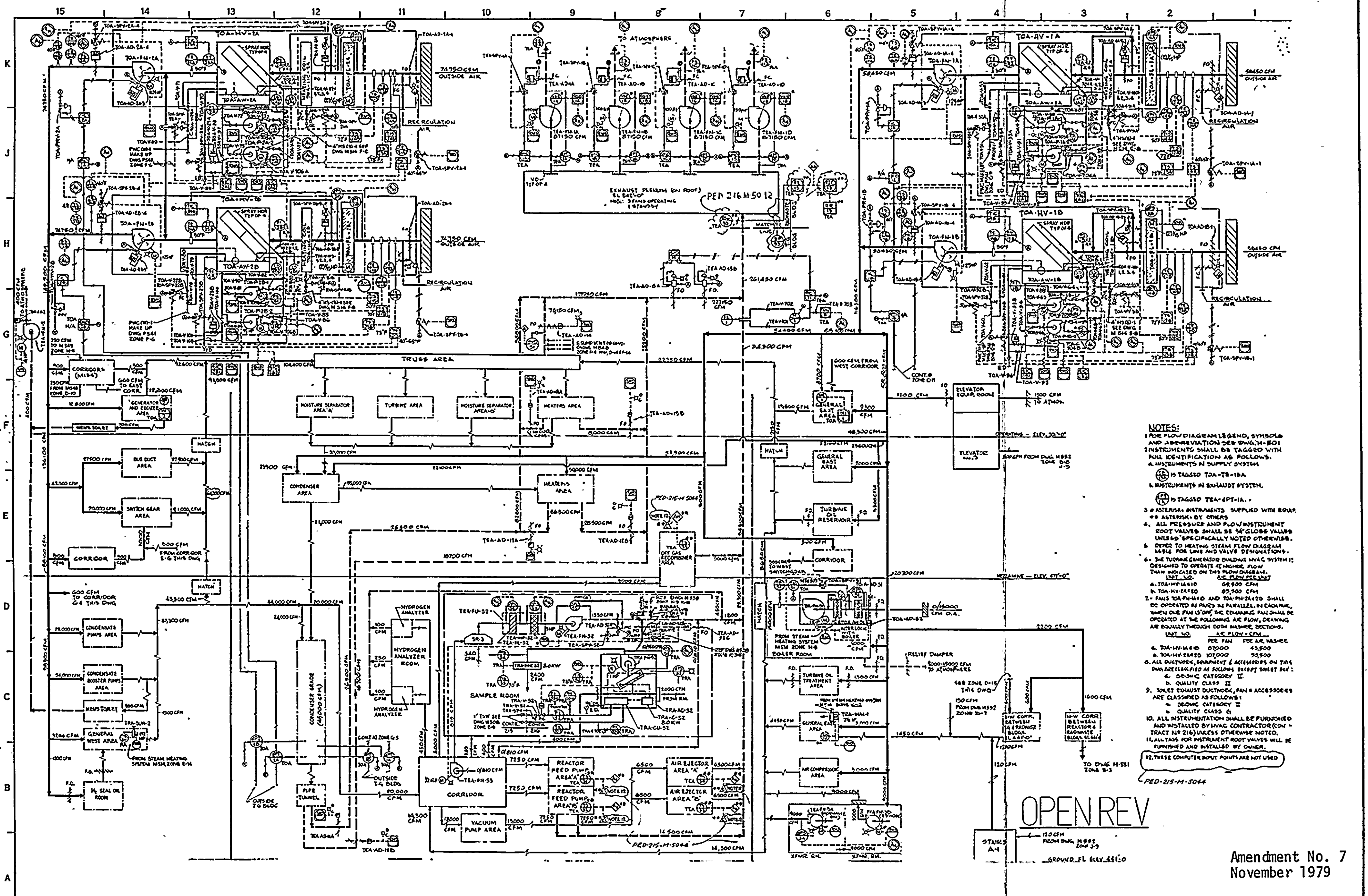
1. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG. M501
2. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE 3/4" GLOBE VALVES UNLESS OTHERWISE NOTED.
3. ALL INSTRUMENTS ON THIS DRAWING TO BE PREFACED BY "WRE" UNLESS OTHERWISE NOTED. I.E. FG 10 IS TAGGED WRE-FG-10
4. ORIFICE PLATE FE-32 SHALL BE SQUARE-EDGED ON BOTH SIDES FOR FLOW IN BOTH DIRECTIONS.
5. ALL DUCTWORK, PIPING, VALVES, EQUIPMENT AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY II
   QUALITY CLASS II
6. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED, AND INSTALLED BY OWNER



WRE-CM-7
NITROGEN CONTROL MANIFOLD
ASSEMBLY DETAIL

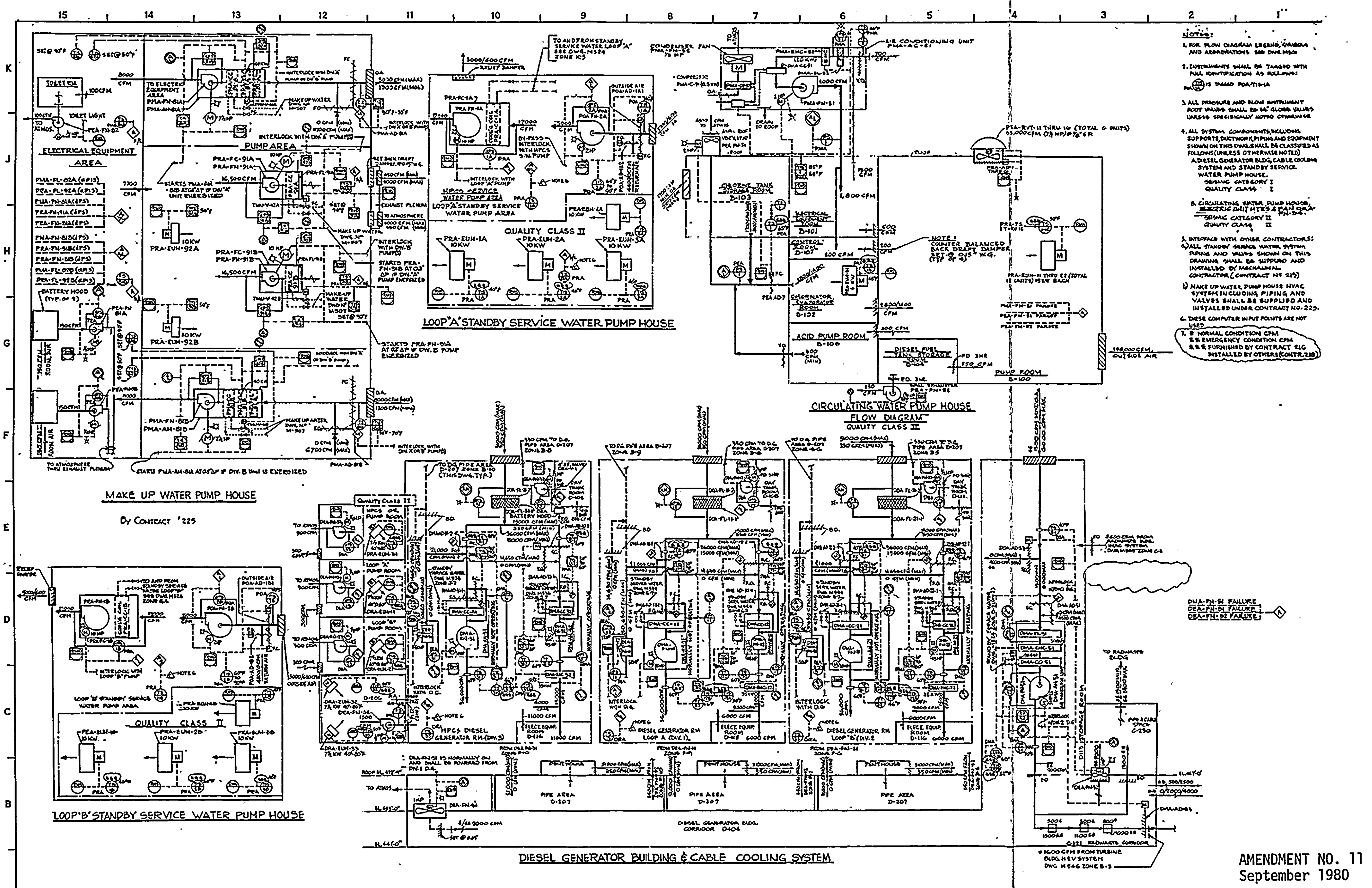
AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC OFF-GAS CHARCOAL ADSORBER VAULT IN RADWASTE BUILDING	FIGURE 9.4-5



Amendment No. 7
November 1979

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC SYSTEMS FOR TURBINE GENERATOR BUILDING	FIGURE 9.4-6



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September 1980

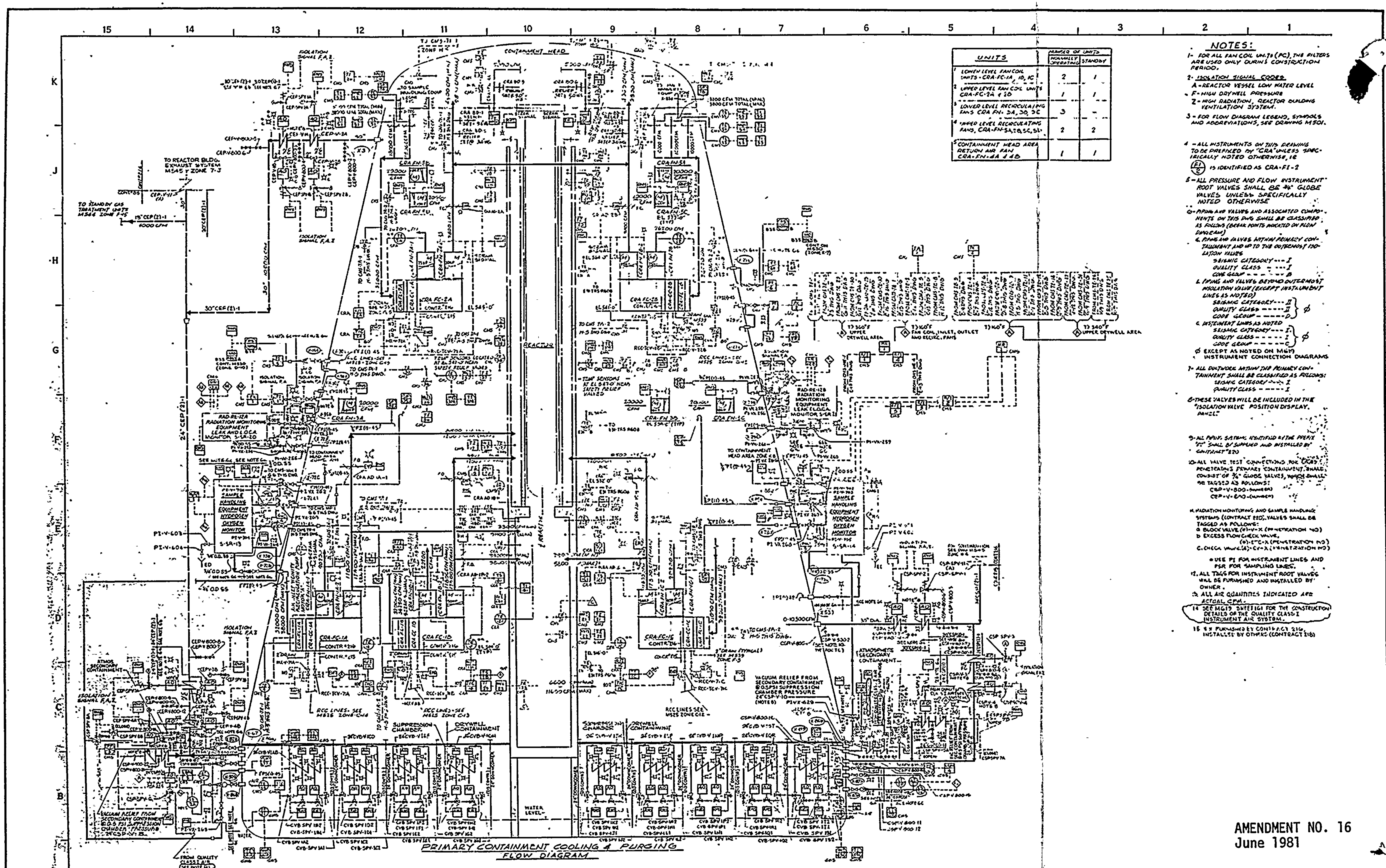
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC - CW, SW & MUW PUMP HOUSES & DIESEL GENERATOR BUILDINGS	FIGURE 9.4-7

DISCLAIMER

FIGURE 9.4-7 (MULTI-COLORED)

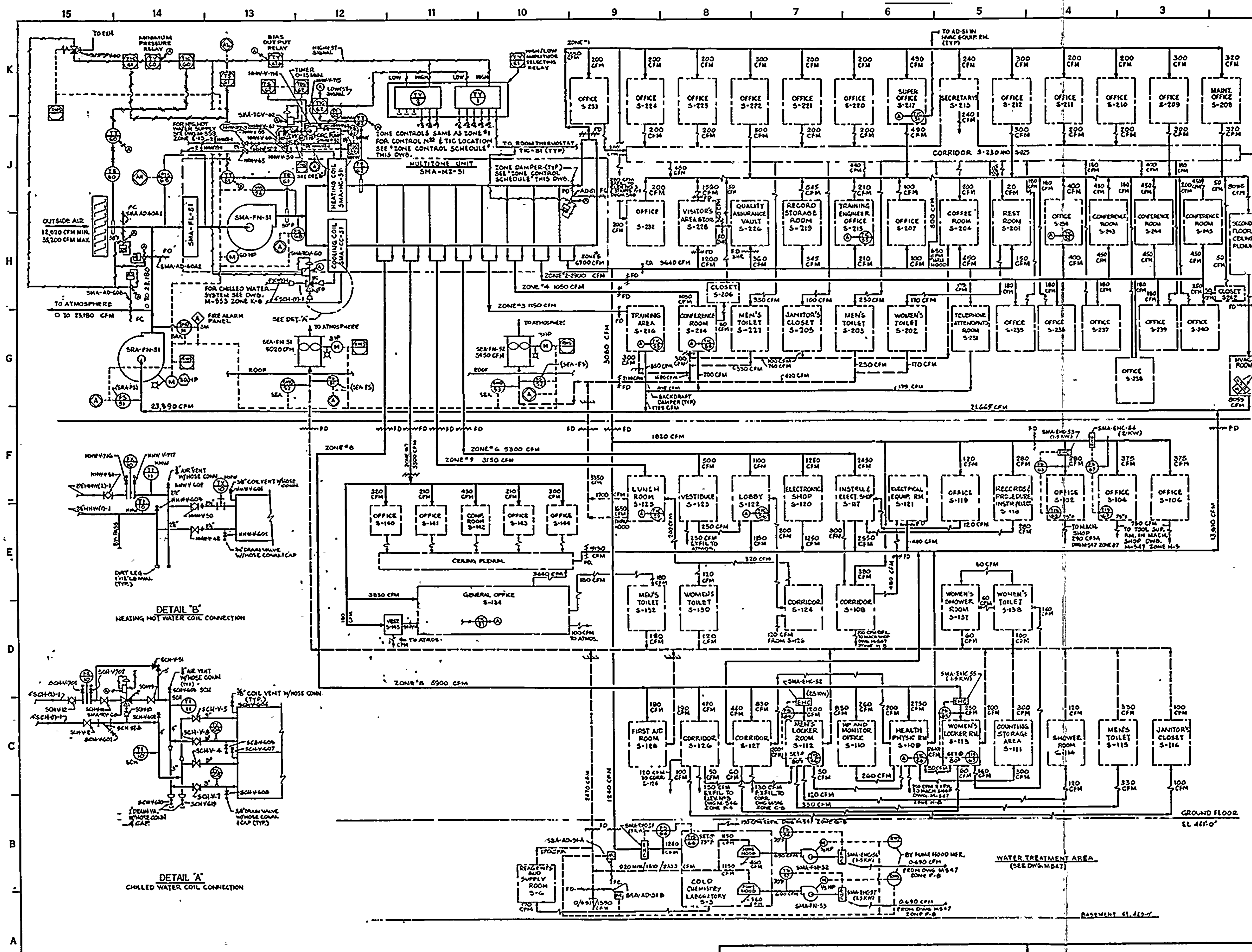
FOR GENERAL SAFETY CLASS INFORMATION ONLY

FOR PROCESS PURPOSES SEE FOLLOWING FIGURE



AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM - REACTOR BUILDING PRIMARY CONTAINMENT COOLING & PURGING SYSTEM	FIGURE 9.4-8



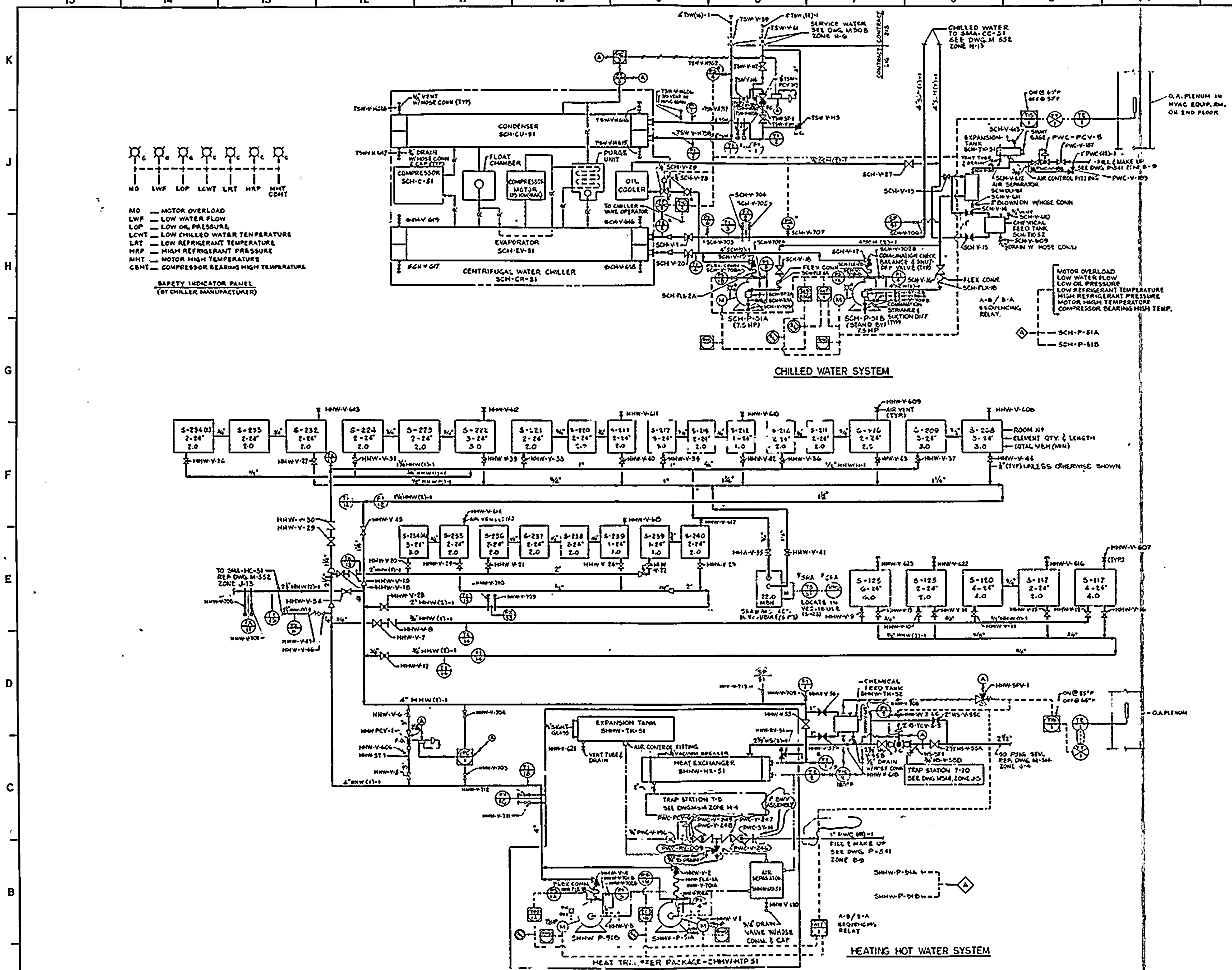
ZONE CONTROL SCHEDULE			
ZONE Nº	TIC LOCATION ROOM Nº	TIC Nº	AD Nº
1	S-217	TIC-51	AD-51
2	S-234	TIC-52	AD-52
3	S-216	TIC-53	AD-53
4	S-214	TIC-54	AD-54
5	S-215	TIC-55	AD-55
6	S-122	TIC-56	AD-56
7	S-134	TIC-57	AD-57
8	S-109	TIC-58	AD-58
9	S-125	TIC-59	AD-59

GENERAL NOTES

1. ALL INSTRUMENTATION ON THIS DWG. TO BE IDENTIFIED BY PREFIX "SMA" UNLESS NOTED OTHERWISE. EX.-SMA-TIC-51
2. ALL INSTRUMENTATION BY HVAC CONTRACTOR. (CONTRACT Nº 216)
3. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG. M-501
4. ALL DUCTWORK, EQUIPMENT AND COMPONENTS SHOWN ON THIS DWG. SHALL BE CLASSIFIED AS FOLLOWS: SEISMIC CATEGORY II QUALITY CLASS "G".
5. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED AND INSTALLED BY OWNER.
6. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE 3/4" GLOBE VALVES UNLESS OTHERWISE NOTED.

AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC SYSTEMS FOR OFFICE AREA AND LABORATORIES IN SERVICE BUILDING	FIGURE 9.4-9

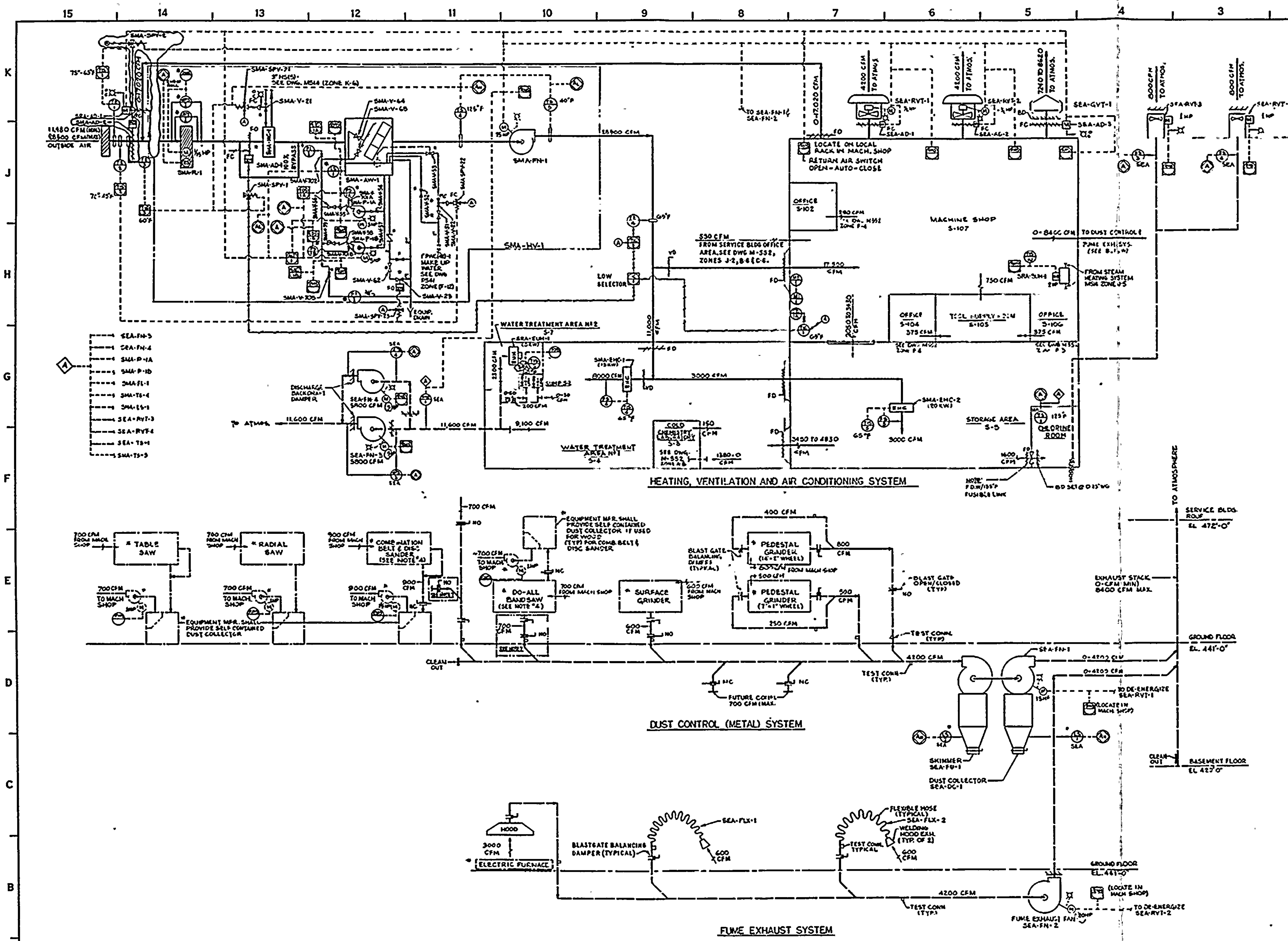


GENERAL NOTES

1. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG. M-501.
2. ALL INSTRUMENTATION FOR CHILLED WATER SYSTEM SHALL BE IDENTIFIED BY PREFIX "SCH", AND ALL INSTRUMENTATION FOR HOT WATER HEATING SYSTEM SHALL BE PREFIXED BY "HHW" UNLESS OTHERWISE NOTED.
3. ALL INSTRUMENTATION SHALL BE BY HVAC CONTRACTOR. (CONTRACT. Nº 216)
4. ALL ITEMS MARKED WITH AN ASTERISK * ARE AVAILABLE WITH ASSOCIATED EQUIPMENT.
5. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE ¾" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
6. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS: SEISMIC CATEGORY II QUALITY CLASS G
7. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED AND INSTALLED BY OWNER.
8. LINES SHOWN (——R——) ARE INTERCONNECTING REFRIGERANT LINES PROVIDED BY CHILLER MANUFACTURER.

AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HEATING HOT WATER AND CHILLED WATER SYSTEMS FOR SERVICE BUILDING	FIGURE 9.4-10



NOTES

1. ALL INSTRUMENTATION ON THIS DRAWING SHALL BE IDENTIFIED BY PREFIX "SMA" UNLESS OTHERWISE NOTED. (EX=SMA-TIC-1).
2. ALL INSTRUMENTATION SHALL BE BY HVAC CONTRACTOR (CONTRACT Nº 216).
3. ALL ITEMS MARKED WITH AN ASTERISK * ARE PROVIDED BY OTHERS.
4. COMBINATION BELT & DISC SANDER AND DO-ALL BANDSAW ARE SHOWN WITH SHUT OFF DAMPERS OPEN TO METAL EXHAUST SYSTEM AND CLOSED TO WOODWORKING EXHAUST SYSTEM. THESE DAMPER POSITIONS MUST BE REVERSED WHEN WORKING WITH WOOD.
5. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS SEE DWG. M-501.
6. ALL DUCTWORK, EQUIPMENT, INSTRUMENTS AND ASSOCIATED COMPONENTS SHOWN ON THIS DWG. SHALL BE CLASSIFIED AS FOLLOWS:

   SEISMIC CATEGORY II
   QUALITY CLASS G
7. DO-ALL BANDSAW & COMBINATION BELT & DISC SANDER DO NOT REQUIRE CONN. FOR MAIN EXHAUST SYSTEM AS PER EQUIPMENT MANUFACTURER. THESE TWO CONN'S WILL BE EXHAUSTING AIR FROM THE MACHINE SHOP AREA, AND WILL BE UTILIZED FOR EXHAUSTING FUTURE MACHINES IF REQUIRED
8. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED AND INSTALLED BY OWNER.
9. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE 3/4" GLOBE VALVES, UNLESS OTHERWISE NOTED

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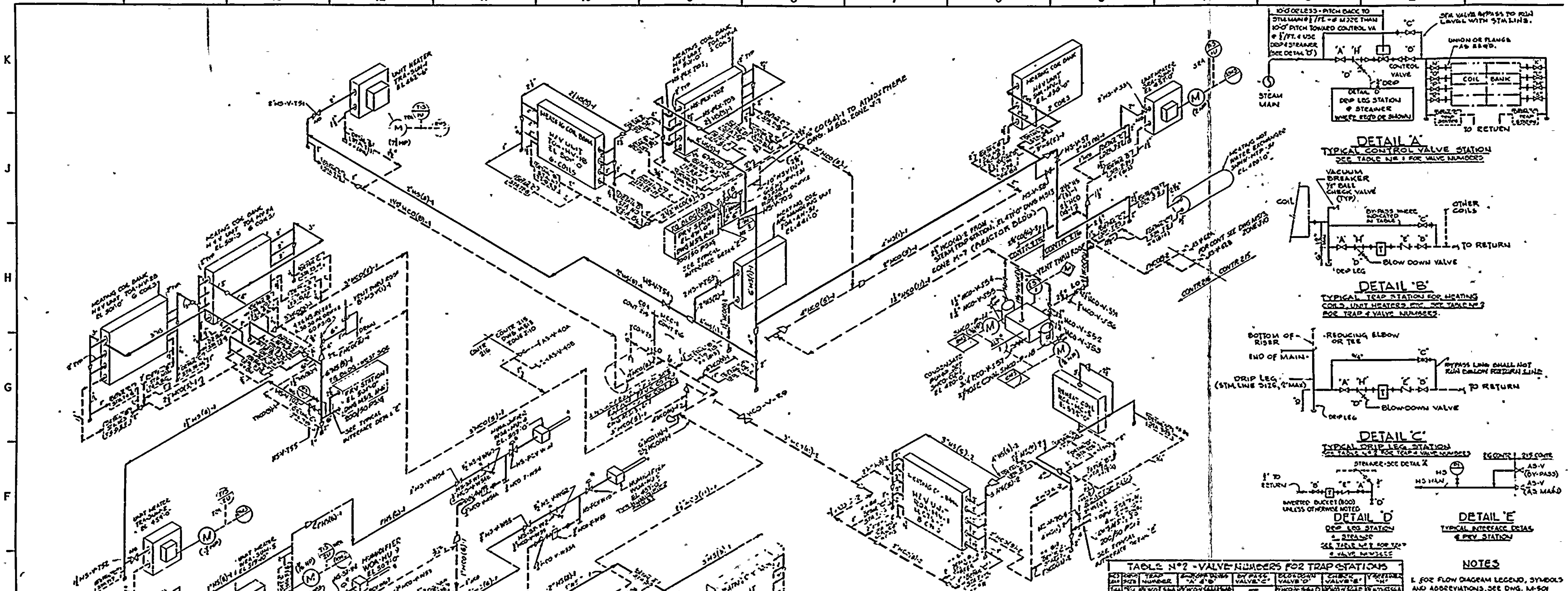
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HVAC SYSTEMS FOR WATER TREATMENT AND MACHINE SHOP AREAS IN SERVICE BUILDING	FIGURE 9.4-11



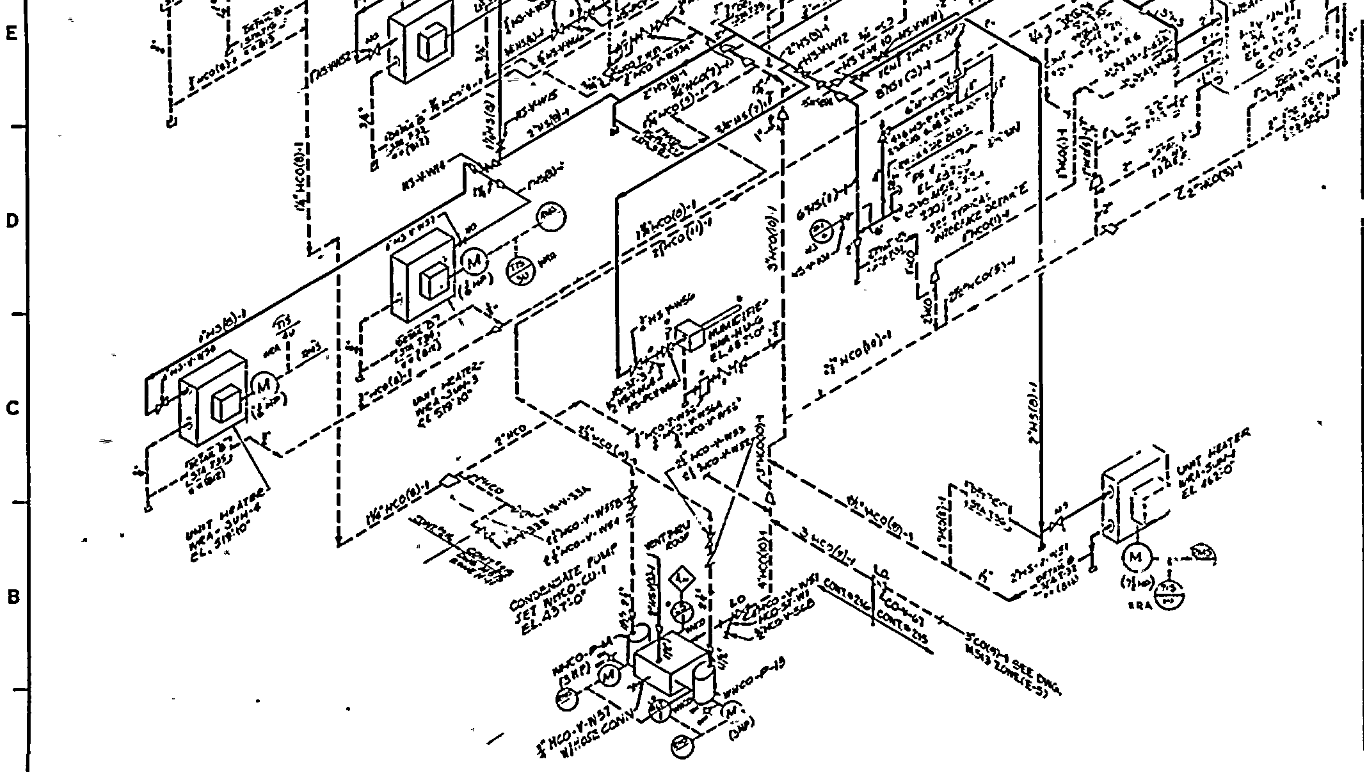


TABLE Nº1 VALVE NUMBERS FOR CONTROL STATIONS

HS STA.#	CONTROL VALVE	SHUT OFF VALVES "A" & "D"	BY-PASS VALVE "C"	BLOWDOWN VALVE "D"	SHUTOFF VALVES K THRU P	Y STRAINER "H"
S-1	3" TOA-V-31A	4" HS-V-T1A & D	3" HS-V-T1C	3/4" HS-V-T1D	2" HS-V-T1K THRU P	4" HCO-ST-S1
S-2	3" TOA-V-31B	4" HS-V-T2A & D	3" HS-V-T2C	3/4" HS-V-T2D	2" HS-V-T2K THRU P	4" HCO-ST-S2
S-3	3" TOA-V-21A	4" HS-V-T3A & D	3" HS-V-T3C	3/4" HS-V-T3D	2" HS-V-T3K THRU P	4" HCO-ST-S3
S-4	3" TOA-V-21D	4" HS-V-T4A & D	3" HS-V-T4C	3/4" HS-V-T4D	2" HS-V-T4K THRU P	4" HCO-ST-S4
S-5	2" HS-ZV-35	2 1/2" HS-V-S5A & D	2" HS-V-S5C	3/4" HS-V-S5D	2 1/2" HS-V-S5K	2" HCO-ST-S5
S-6	3" FAA-V-1	3" HS-V-S6A & D	1 1/2" HS-V-S6C	3/4" HS-V-S6D	2" HS-V-S6K & L	3" HCO-ST-S6
S-7	3" ROA-V-21	4" HS-V-R7A & D	3" HS-V-R7C	3/4" HS-V-R7D	2" HS-V-R7K THRU P	4" HCO-ST-S7
S-8	2" ROA-V-31	2 1/2" HS-V-R8A & D	2" HS-V-R8C	3/4" HS-V-R8D	1 1/2" HS-V-R8K	2 1/2" HCO-ST-S8
S-9	4" WOA-V-21	4" HS-V-W9A & D	3" HS-V-W9C	3/4" HS-V-W9D	2" HS-V-W9K THRU P	4" HCO-ST-S9

TABLE Nº2 VALVE NUMBERS FOR TRAP STATIONS

HCO STA.#	ORIFICE SIZE	TRAP NUMBER	SHUTOFF VALVES "A" & "B"	BY-PASS VALVE "C"	BLOWDOWN VALVE "D"	CHECK VALVE "E"	Y STRAINER "H"
T-1A	3/64	1 1/2" HCO-T-T1A	1 1/2" HCO-V-T1AA & T1AB	—	1/2" HCO-V-T1AD	1 1/2" HCO-V-T1AE	1 1/2" HCO-ST-T1A
T-1B	13/64	1 1/2" HCO-T-T1B	1 1/2" HCO-V-T1BA & T1BB	—	1/2" HCO-V-T1BD	1 1/2" HCO-V-T1BE	1 1/2" HCO-ST-T1B
T-1C	17/64	1 1/2" HCO-T-T1C	1 1/2" HCO-V-T1CA & T1CB	—	1/2" HCO-V-T1CD	1 1/2" HCO-V-T1CE	1 1/2" HCO-ST-T1C
T-1D	13/64	1 1/2" HCO-T-T1D	1 1/2" HCO-V-T1DA & T1DB	—	1/2" HCO-V-T1DD	1 1/2" HCO-V-T1DE	1 1/2" HCO-ST-T1D
T-1E	13/64	1 1/2" HCO-T-T1E	1 1/2" HCO-V-T1EA & T1EB	—	1/2" HCO-V-T1ED	1 1/2" HCO-V-T1EE	1 1/2" HCO-ST-T1E
T-1F	13/64	1 1/2" HCO-T-T1F	1 1/2" HCO-V-T1FA & T1FB	—	1/2" HCO-V-T1FD	1 1/2" HCO-V-T1FE	1 1/2" HCO-ST-T1F
T-1G	11/64	1" HCO-T-T1G	1" HCO-V-T1GA & T1GB	1/2" HCO-V-T1GC	1/2" HCO-V-T1GD	1" HCO-V-T1GE	1" HCO-ST-T1G
T-1H	11/64	1" HCO-T-T1H	1" HCO-V-T1HA & T1HB	1/2" HCO-V-T1HC	1/2" HCO-V-T1HD	1" HCO-V-T1HE	1" HCO-ST-T1H
T-2A	13/64	1 1/2" HCO-T-T2A	1 1/2" HCO-V-T2AA & T2AB	—	1/2" HCO-V-T2AD	1 1/2" HCO-V-T2AE	1 1/2" HCO-ST-T2A
T-2B	13/64	1 1/2" HCO-T-T2B	1 1/2" HCO-V-T2BA & T2BB	—	1/2" HCO-V-T2BD	1 1/2" HCO-V-T2BE	1 1/2" HCO-ST-T2B
T-2C	13/64	1 1/2" HCO-T-T2C	1 1/2" HCO-V-T2CA & T2CB	—	1/2" HCO-V-T2CD	1 1/2" HCO-V-T2CE	1 1/2" HCO-ST-T2C
T-2D	13/64	1 1/2" HCO-T-T2D	1 1/2" HCO-V-T2DA & T2DB	—	1/2" HCO-V-T2DD	1 1/2" HCO-V-T2DE	1 1/2" HCO-ST-T2D
T-2E	13/64	1 1/2" HCO-T-T2E	1 1/2" HCO-V-T2EA & T2EB	—	1/2" HCO-V-T2ED	1 1/2" HCO-V-T2EE	1 1/2" HCO-ST-T2E
T-2F	13/64	1 1/2" HCO-T-T2F	1 1/2" HCO-V-T2FA & T2FB	—	1/2" HCO-V-T2FD	1 1/2" HCO-V-T2FE	1 1/2" HCO-ST-T2F
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

TABLE Nº2 - VALVE NUMBERS FOR TRAP STATIONS

HCO STA.#	ORIFICE SIZE	TRAP NUMBER	SHUTOFF VALVES "A" & "B"	BY-PASS VALVE "C"	BLOWDOWN VALVE "D"	CHECK VALVE "E"	Y STRAINER "H"
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

NOTES

1. FOR FLOW DIAGRAM LEGEND, SYMBOLS AND ABBREVIATIONS, SEE DWG. M-501
2. ALL PRESSURE AND FLOW INSTRUMENT ROOT VALVES SHALL BE 3/4" GLOBE VALVES UNLESS OTHERWISE NOTED
3. ALL ITEMS MARKED WITH AN ASTERISK (*) ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
4. ALL TRAPS SHALL BE FLOAT AND THERMOSTATIC TYPE UNLESS OTHERWISE NOTED
5. ALL TRAPS MARKED WITH A DOUBLE ASTERISK (**) SHALL BE INVERTED (OPEN) BUCKET TYPE (WITH THERMIC VENT BUCKET) WITH ARMSTRONG NUMBER AS NOTED OR APPROVED EQUAL.
6. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DWG. UNLESS OTHERWISE NOTED SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS - II
   SEISMIC CATEGORY - II
   CODE GROUP - D
   ALL PIPING, VALVES & ASSOCIATED COMPONENTS IN REACTOR BLDG SHALL BE CLASSIFIED AS FOLLOWS: (SEE CONTR. 216 SPEC. SECTIONS 15A & 15E)
   a) SEISMIC CATEGORY:
      PIPING & PIPING SUPPORTS - I
      VALVES & ASSOC. COMPONENTS - II
   b) QUALITY CLASS:
      PIPING SUPPORTS - I
      PIPING, VALVES & ASSOC. COMPONENTS II
7. ALL TAGS FOR INSTRUMENT ROOT VALVES WILL BE FURNISHED & INSTALLED BY OWNER

SYMBOLS

——— DENOTES STEAM

– – – DENOTES CONDENSATE

FOR OTHER SYMBOLS SEE DWG M501

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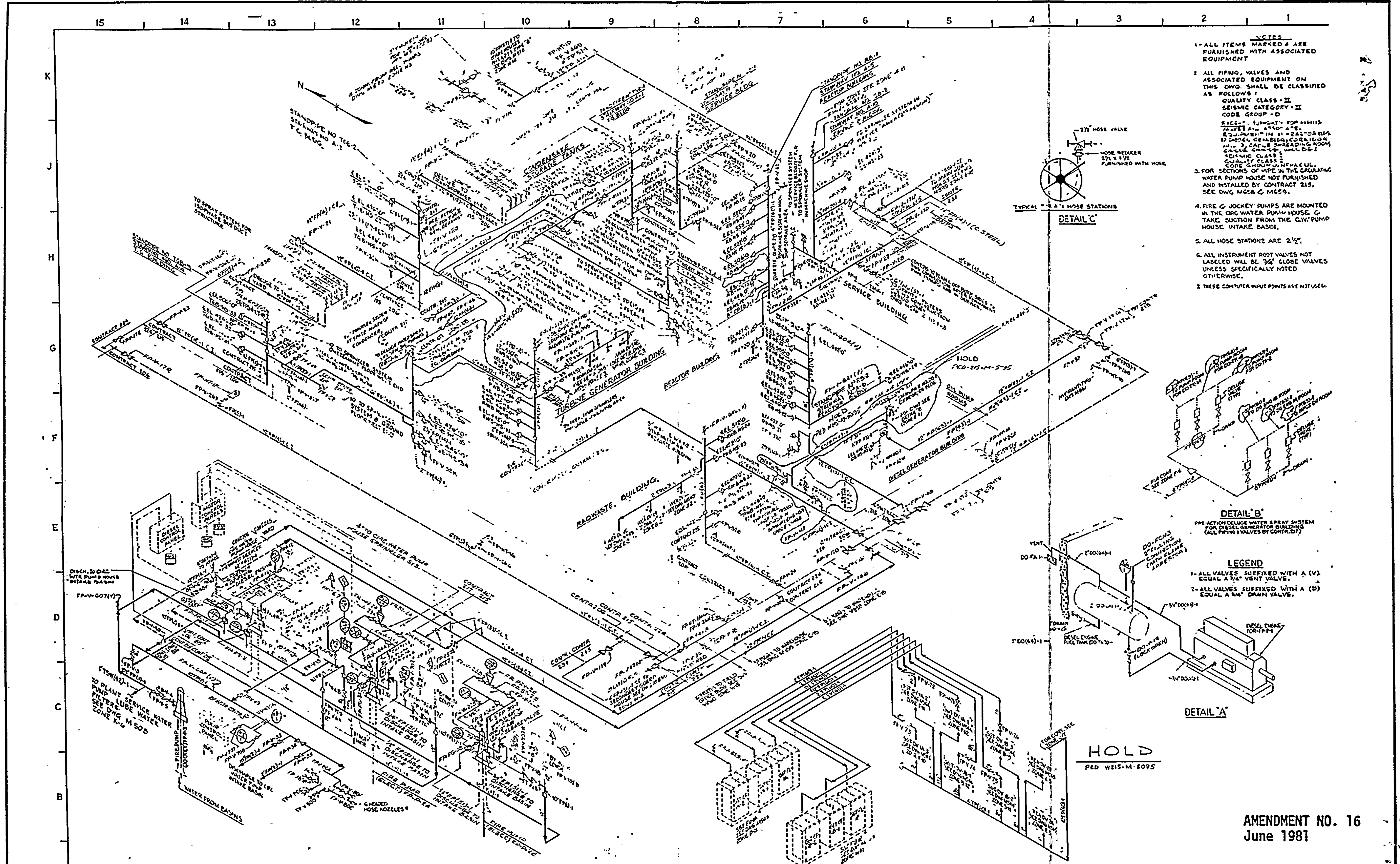
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HEATING STEAM SYSTEM ALL BUILDINGS	FIGURE 9.4-12



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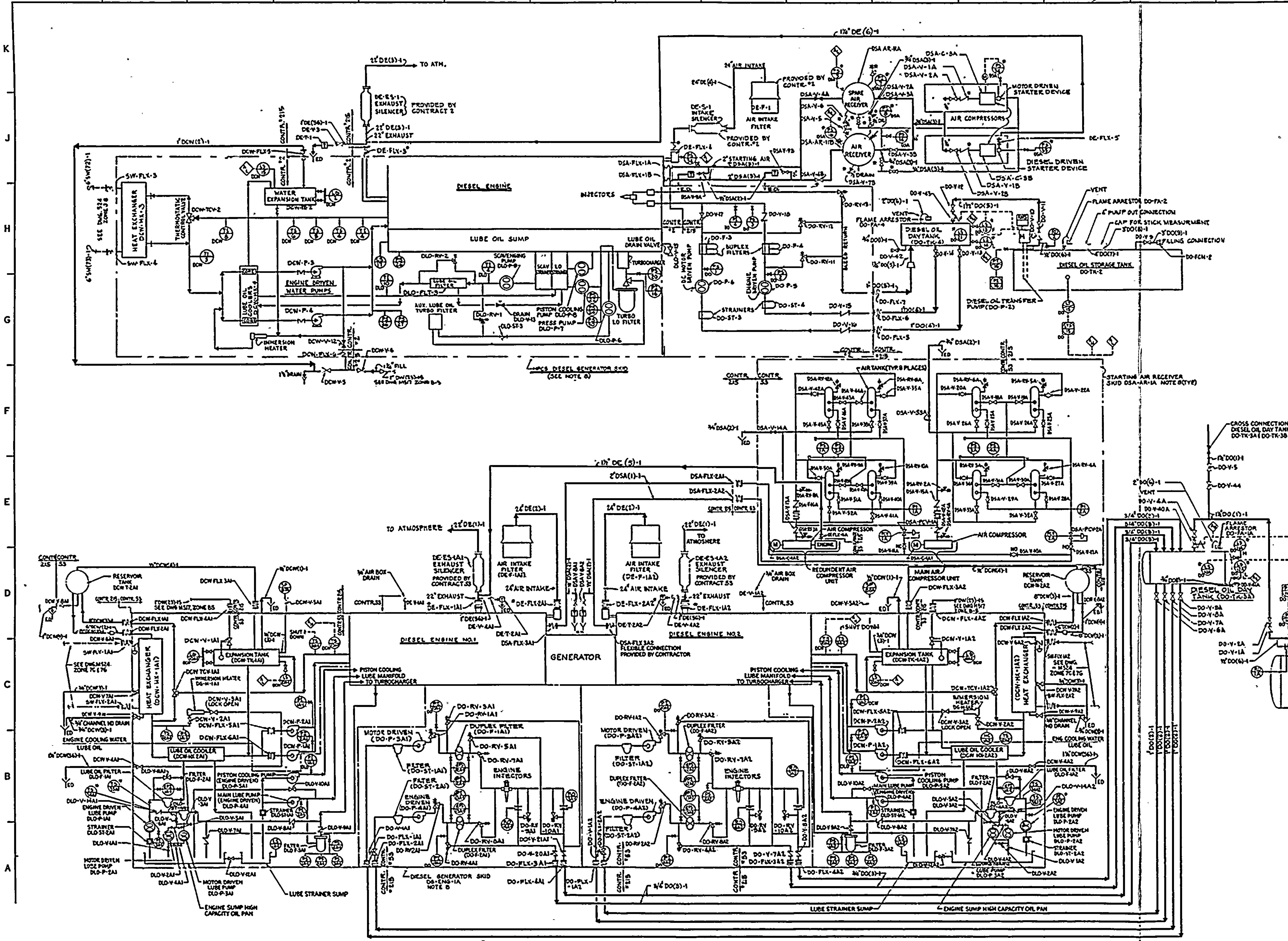
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FIRE PROTECTION SYSTEM	FIGURE 9.5-1



NOTES

1. ALL DSA, DCW & DE PIPING AND ASSOCIATED COMPONENTS ON THIS DWG INCLUDED IN CONTRACT 215 ARE CLASSIFIED.

   QUALITY CLASS I
   CODE GROUP D
   SEISMIC CATEGORY I

2. ALL DIESEL OIL PIPING AND ASSOCIATED COMPONENTS ON THIS DWG INCLUDED IN CONTRACT 215 ARE CLASSIFIED.

   QUALITY CLASS I
   CODE GROUP C
   SEISMIC CATEGORY I

3. ALL PIPING VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING INCLUDED IN CONTRACT 215 EXCEPT WHERE CONTRACT INTERFACES ARE DENOTED BY

   CONTRACT #215
   CONTRACT #2 OR #53

4. ALL PIPING AND FITTINGS FOR DE(1)-1 AND DE(2)-1 ARE SUPPLIED BY CONTRACT #53 AND SUPPORTED AND ERECTED BY CONTRACT #215

5. PIPING AND EQUIPMENT FOR DIESEL GENERATOR, DG-ENG-1B IS IDENTICAL TO THAT SHOWN FOR DIESEL GENERATOR, DG-ENG-1A

6. EQUIPMENT TAG NUMBER WITH SUFFIX A1 REFERS TO NO.1 ENGINE OF DIESEL GENERATOR DG ENG-1A. SUFFIX A2 REFERS TO NO.2 ENGINE OF DIESEL GENERATOR DG-ENG-1A. SUFFIX B1 REFERS TO NO.1 ENGINE OF DIESEL GENERATOR DG-ENG-1B AND SUFFIX B2 REFERS TO NO.2 ENGINE OF DIESEL GENERATOR DG-ENG-1B

7. ALL ITEMS MARKED # ARE FURNISHED WITH ASSOCIATED EQUIPMENT.

8. ALL PIPING, COMPONENTS, AND INSTRUMENTATION SHOWN WITHIN THE CONFINES OF THE SKID PACKAGE ARE SUPPLIED WITH THE ASSOCIATED EQUIPMENT.

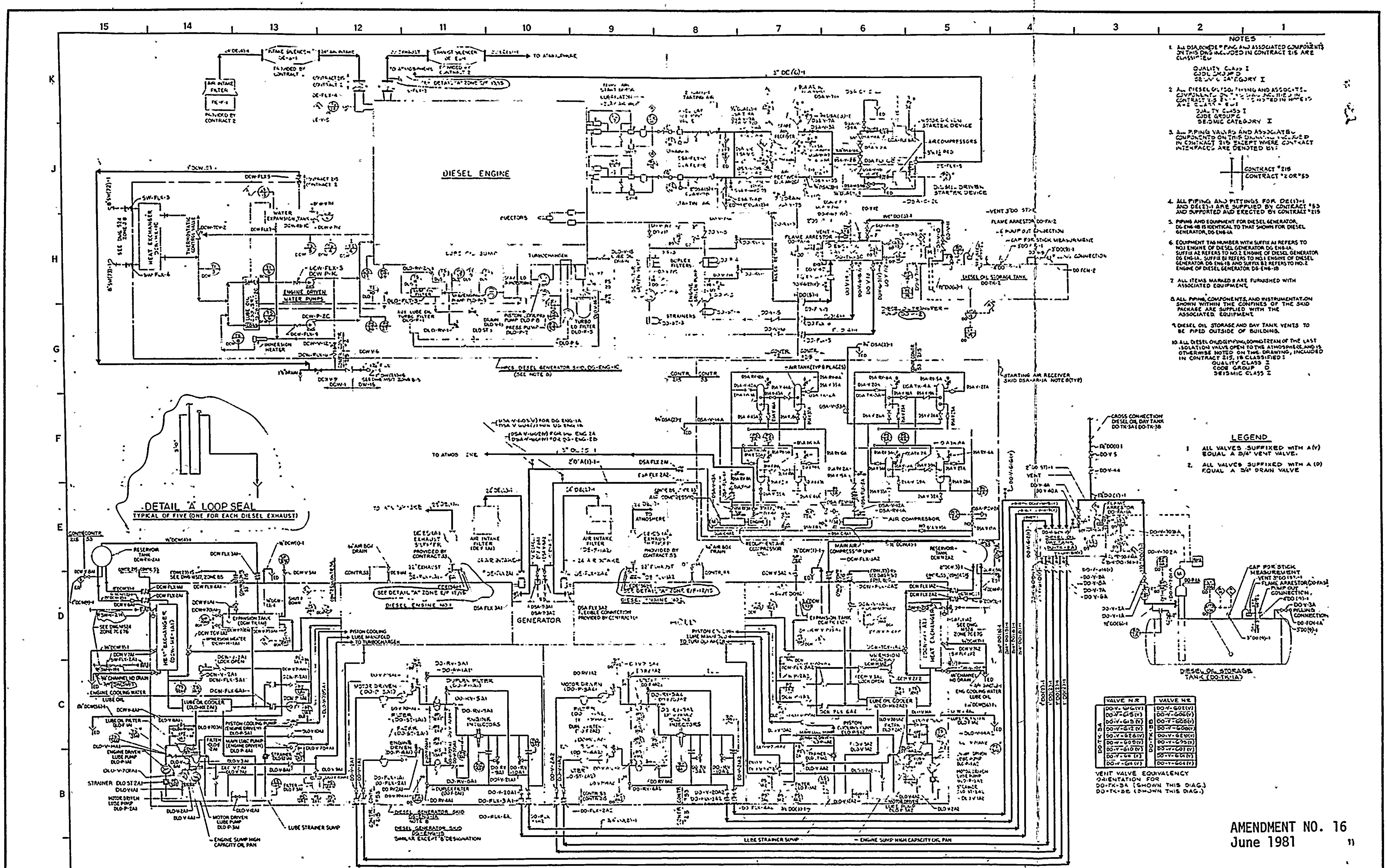
9. DIESEL OIL STORAGE AND DAY TANK VENTS TO BE PIPED OUTSIDE OF BUILDING.

Legend
Safety Class
1.
2.
3.
G.

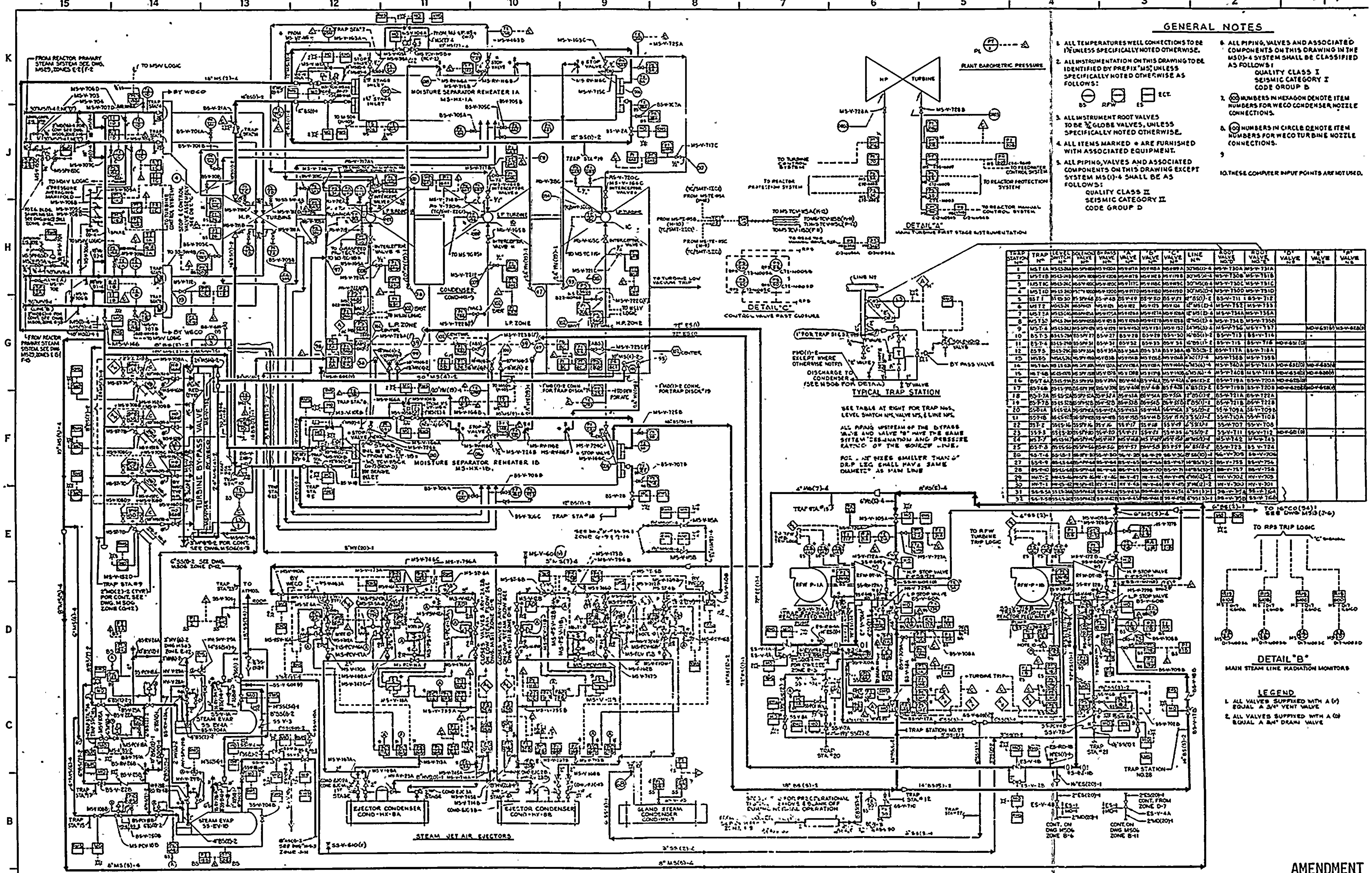
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

DIESEL OIL AND MISC. SYSTEMS

FIGURE 9.5-4



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	DIESEL OIL AND MISC. SYSTEMS	FIGURE 9.5-4



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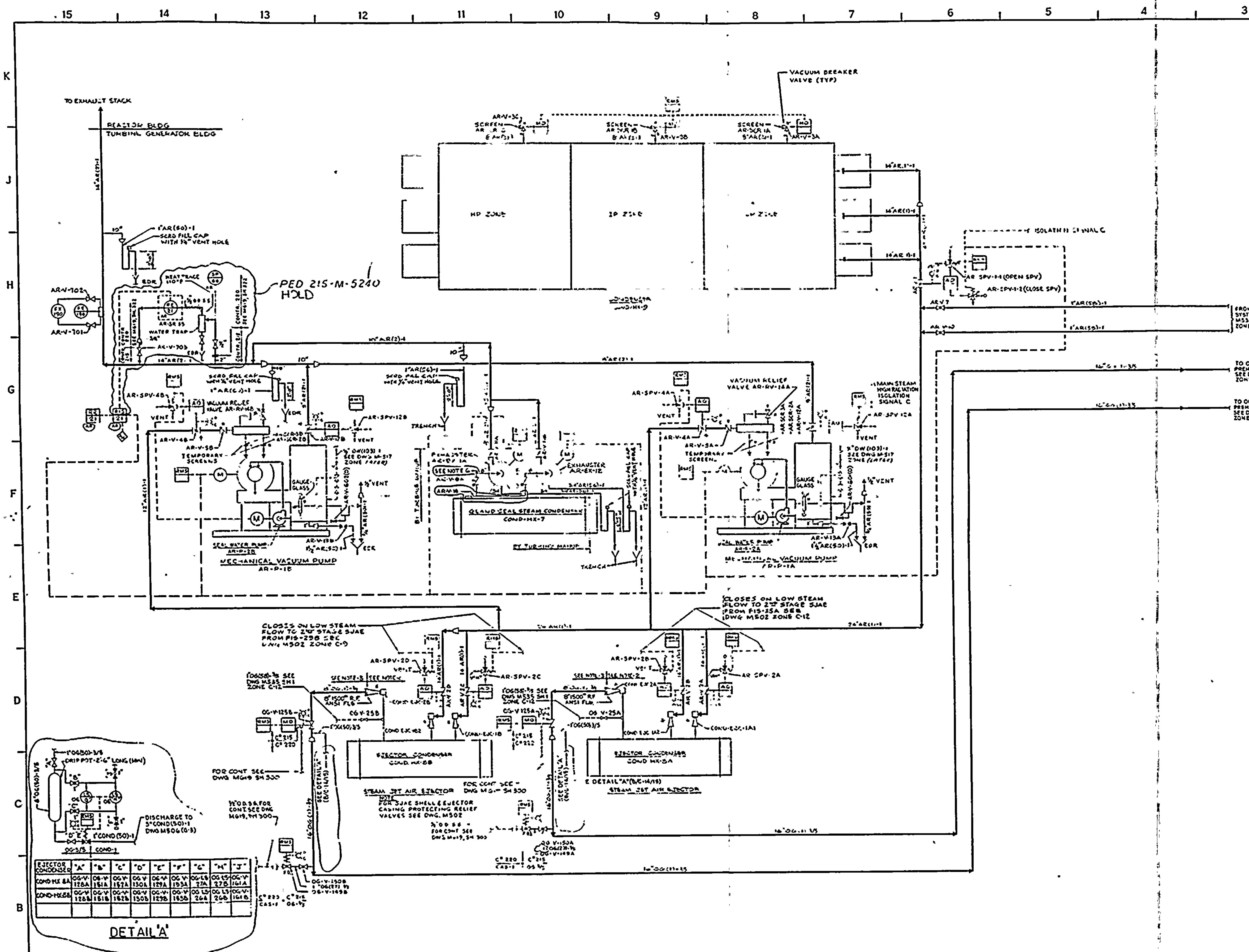
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	MAIN STEAM SUPPLY SYSTEM	FIGURE 10.3-1



NOTES

1. ALL ITEM MARKED ▸ ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
2. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING IN THE AIR REMOVAL SYSTEM SHALL BE AS FOLLOWS EXCEPT FOR THE AIR EJECTOR DISCHARGE'S.
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
3. ALL PIPING, VALVES & ASSOCIATED COMPONENTS ON THIS DRAWING IN THE OFF GAS SYSTEM OR AIR EJECTOR DISCHARGE'S SHALL BE AS FOLLOWS.
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP C
5. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
6. GLAND STEAM CONDENSER EXHAUSTER (AR-EX-1A(1B) SUCTION PIPING INCLUDING CHECK VALVES AR-V-8A&8B ARE FURNISHED WITH THE GLAND STEAM CONDENSER (CONTRACT #1) WITH THE EXCEPTION THAT THROTTLE VALVE AR-V-18 AND ITS MATING SET OF FLANGES ARE TO BE FURNISHED AND INSTALLED BY THE 215 CONTRACTOR IN THE LOCATION SHOWN.

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A 3/4" DRAIN VALVE.

HOLD
PED-215-M-5240

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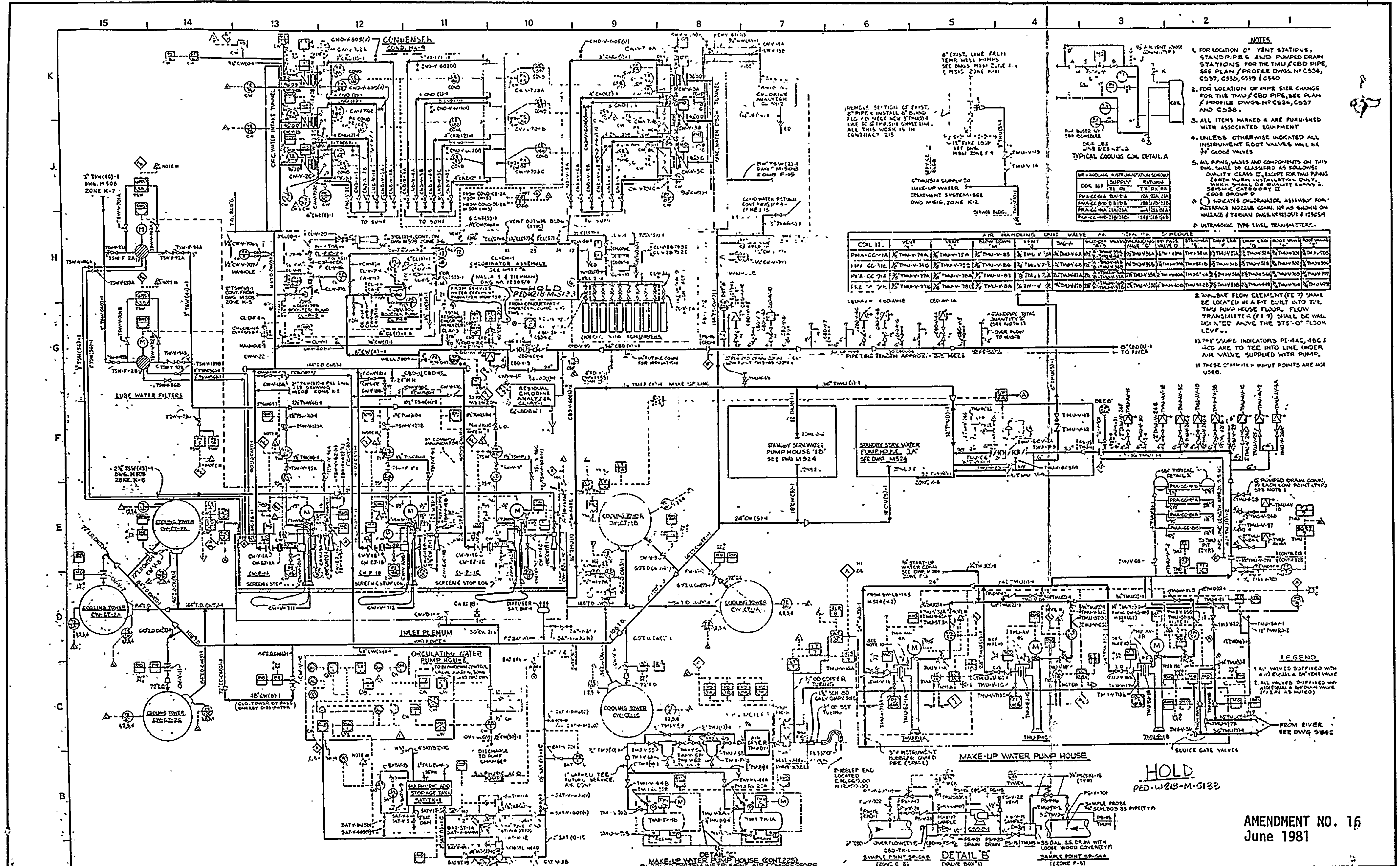
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM GAS & AIR REMOVAL SYSTEM TURBINE GENERATOR BLDG.	FIGURE 10.4-1



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM CIRCULATING WATER SYS. TURBINE GENERATOR BLDG. & YARD	FIGURE 10.4-3

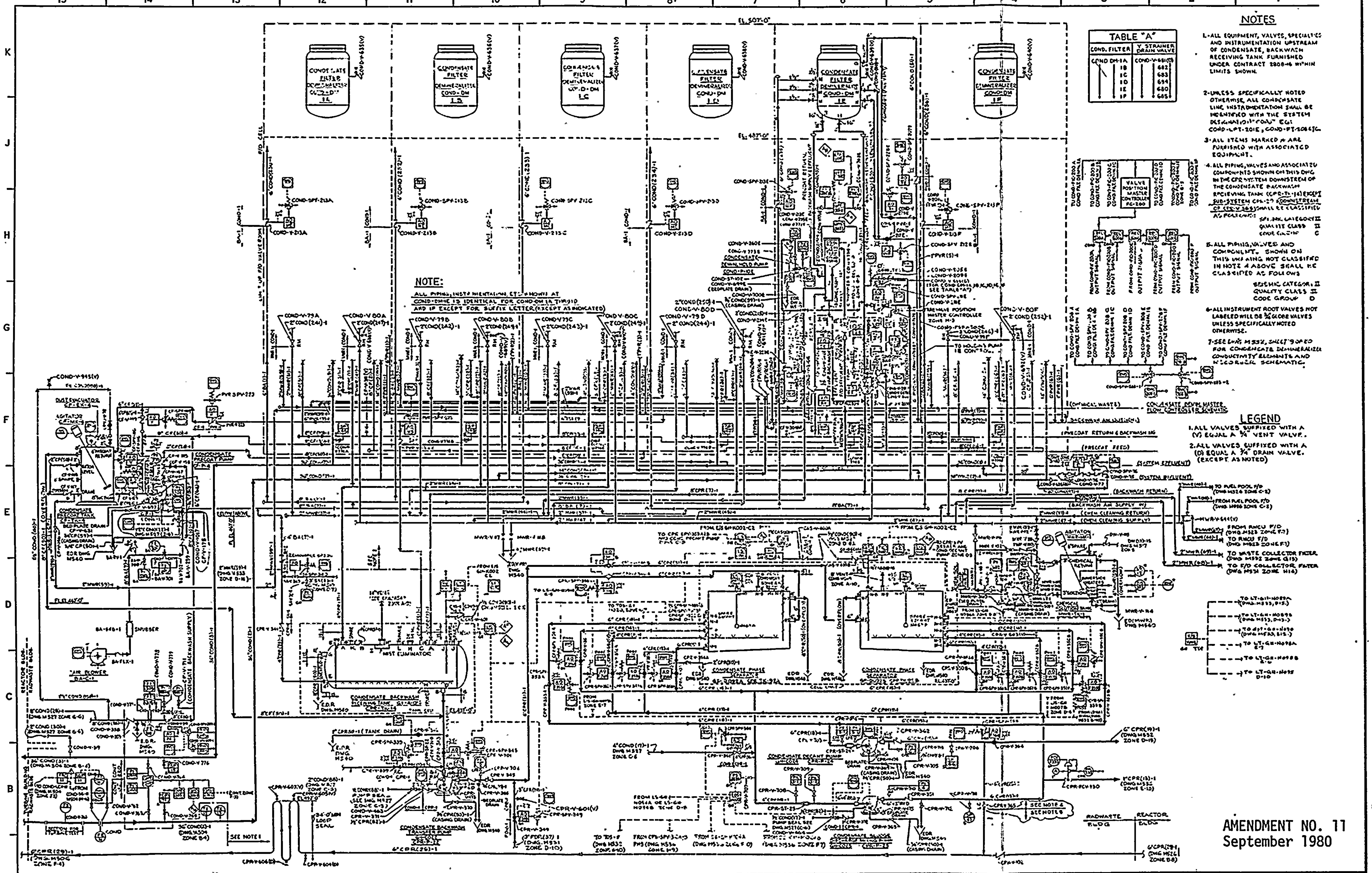


TABLE "A"

COND. FILTER	Y STRAINER DRAIN VALVE
COND DM-1A	COND-V-681(D)
1B	682
1C	683
1D	694
1E	680
1F	685

NOTES

1. ALL EQUIPMENT, VALVES, SPECIALTIES AND INSTRUMENTATION UPSTREAM OF CONDENSATE BACKWASH RECEIVING TANK FURNISHED UNDER CONTRACT 2808-46 WITHIN LIMITS SHOWN.
2. UNLESS SPECIFICALLY NOTED OTHERWISE ALL CONDENSATE LINE INSTRUMENTATION SHALL BE IDENTIFIED WITH THE SYSTEM DESIGNATION "COND" EG: COND-LPT-201E, COND-FT-206A/C.
3. ALL ITEMS MARKED ✱ ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
4. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS SHOWN ON THIS DWG IN THE CPR SYSTEM DOWNSTREAM OF THE CONDENSATE BACKWASH RECIEVING TANK (CPR-TK-14) EXCEPT SUB-SYSTEM CPR-29 (DOWNSTREAM OF CPR-V-365) SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY II
   QUALITY CLASS II
   CODE GROUP C
5. ALL PIPING, VALVES AND COMPONENTS SHOWN ON THIS DRAWING NOT CLASSIFIED IN NOTE 4 ABOVE SHALL BE CLASSIFIED AS FOLLOWS
   SEISMIC CATEGORY II
   QUALITY CLASS II
   CODE GROUP D
6. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4 GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
7. SEE DWG M552, SHEET 9 OF 10 FOR CONDENSATE DEMINERALIZER CONDUCTIVITY ELEMENTS AND RECORDER SCHEMATIC.

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A 3/4" DRAIN VALVE. (EXCEPT AS NOTED)

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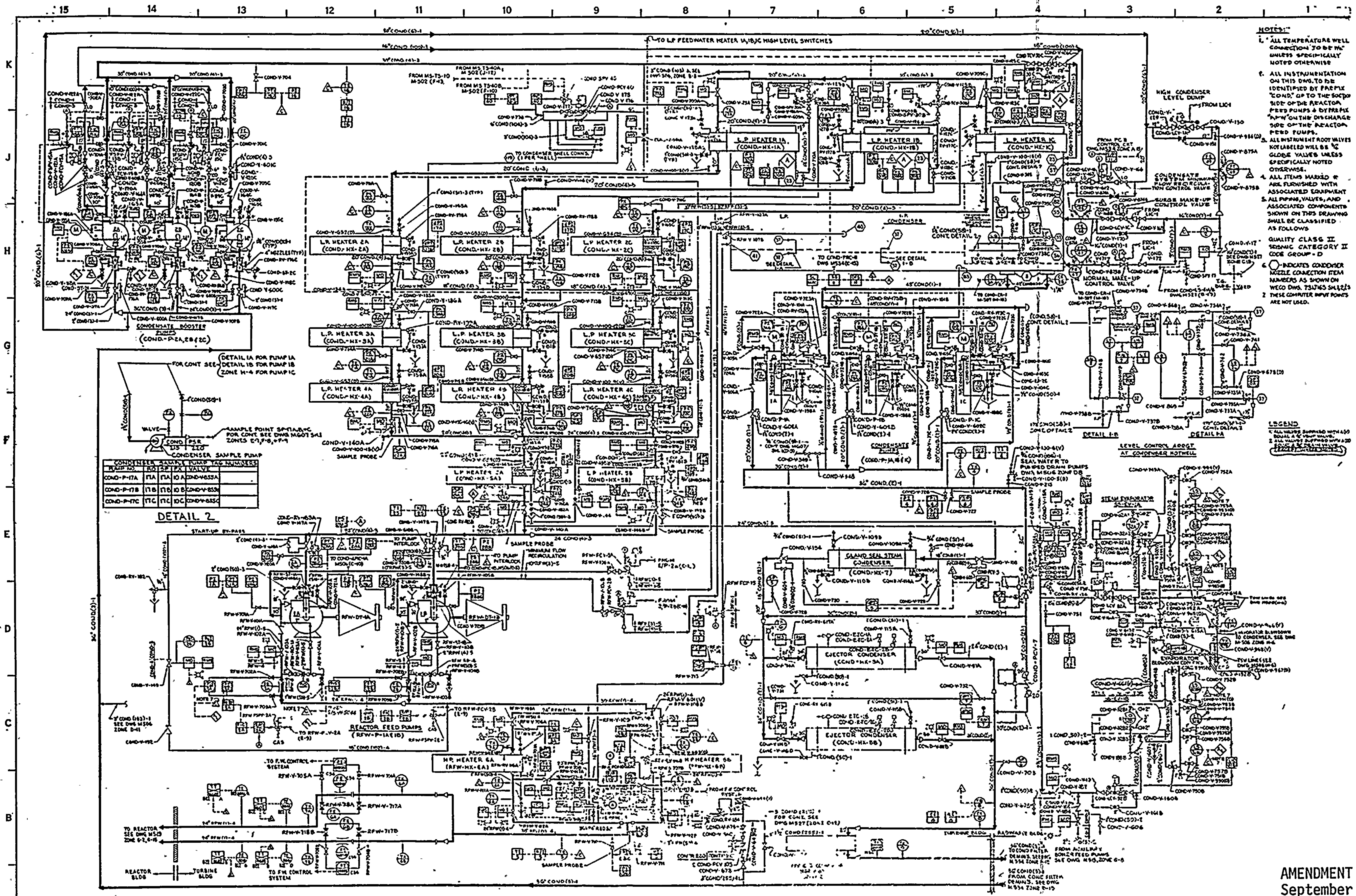
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM RADIOACTIVE WASTE SYSTEM CONDENSATE DEMINERALIZATION	FIGURE 10.4-4



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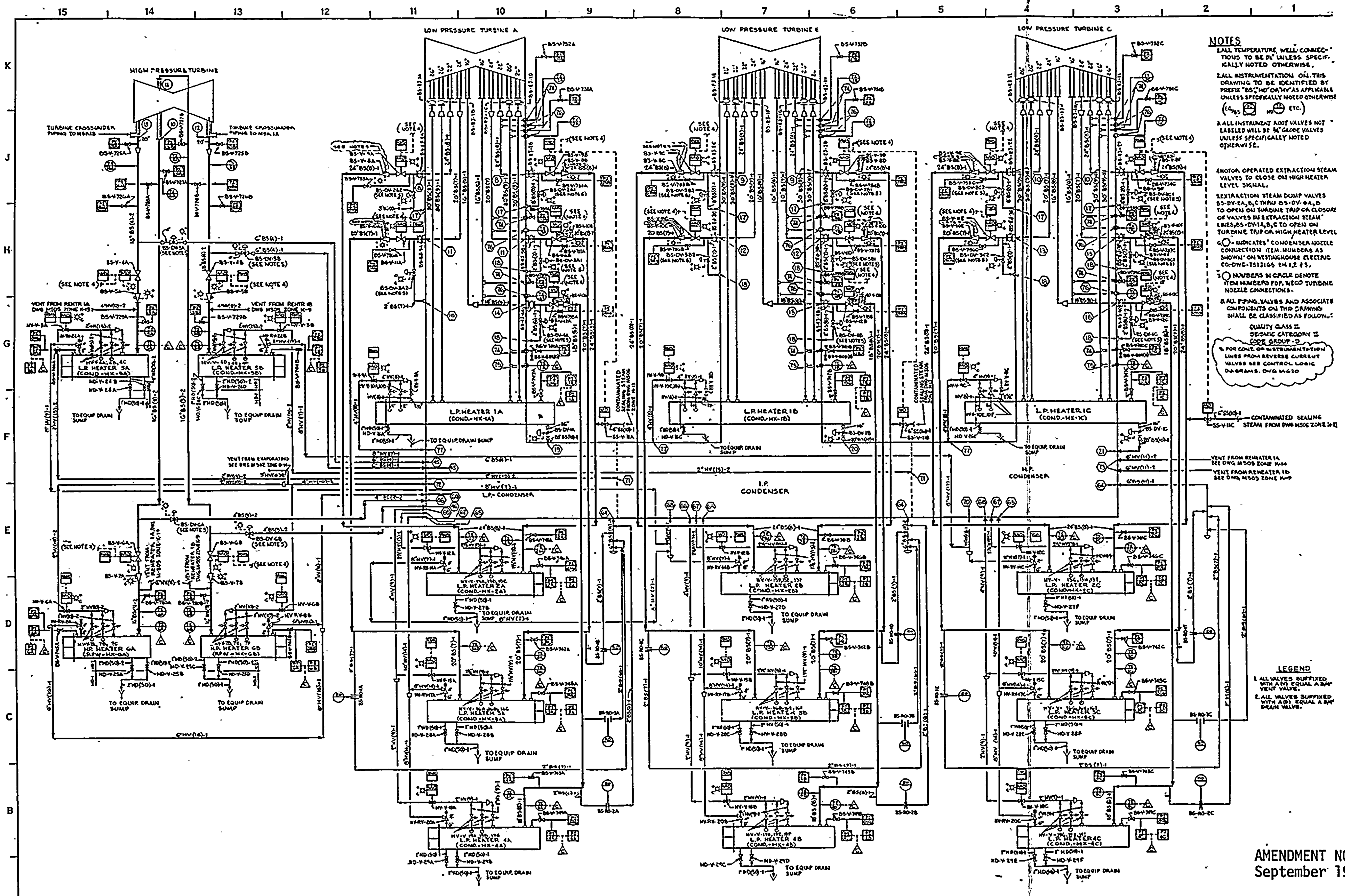
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM CONDENSATE & FEED WATER SYSTEM	FIGURE 10.4-5



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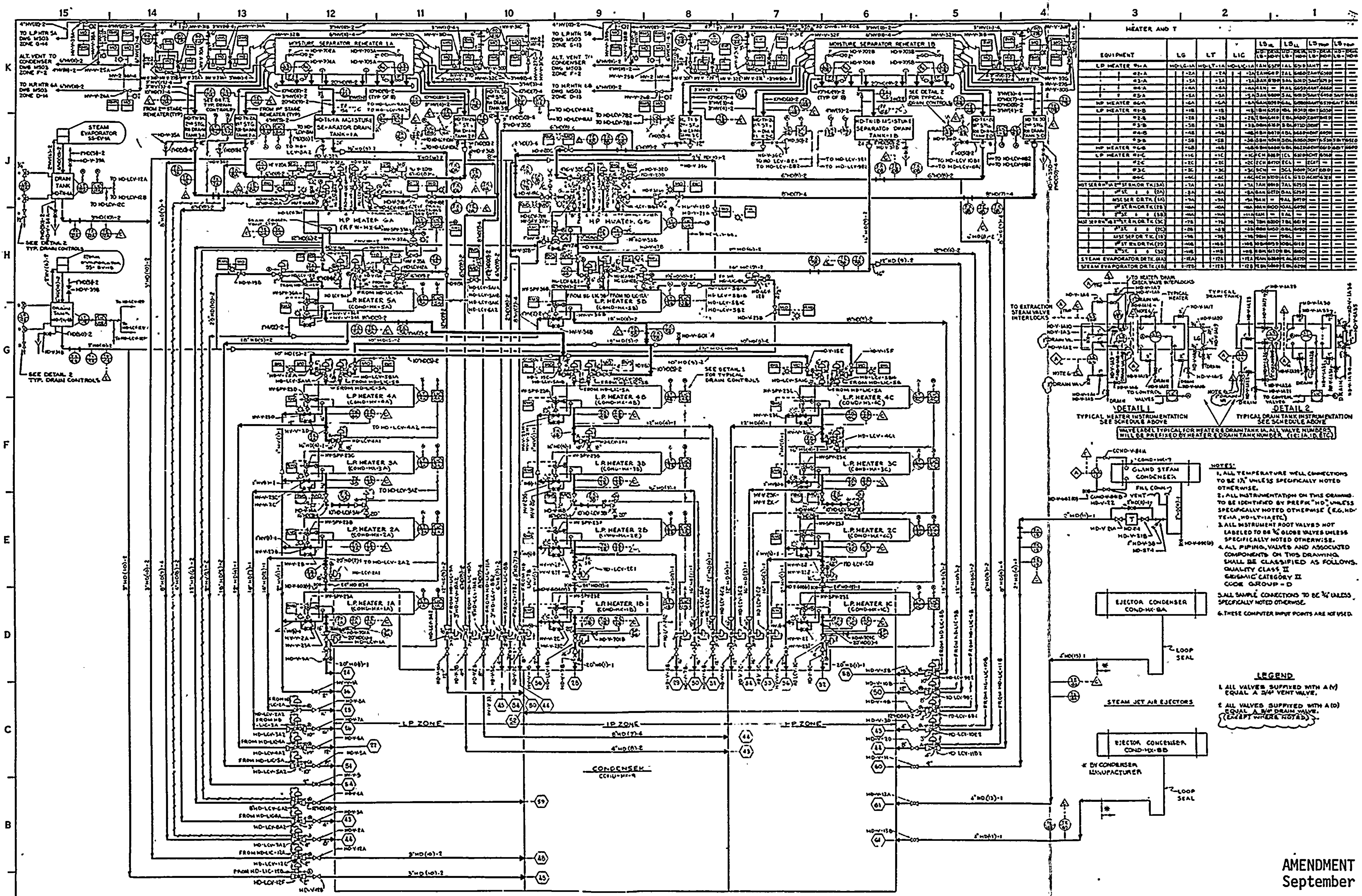
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM EXTRACTION STEAM & HEATER VENTS TURBINE GENERATOR BLDG.	FIGURE 10.4-6



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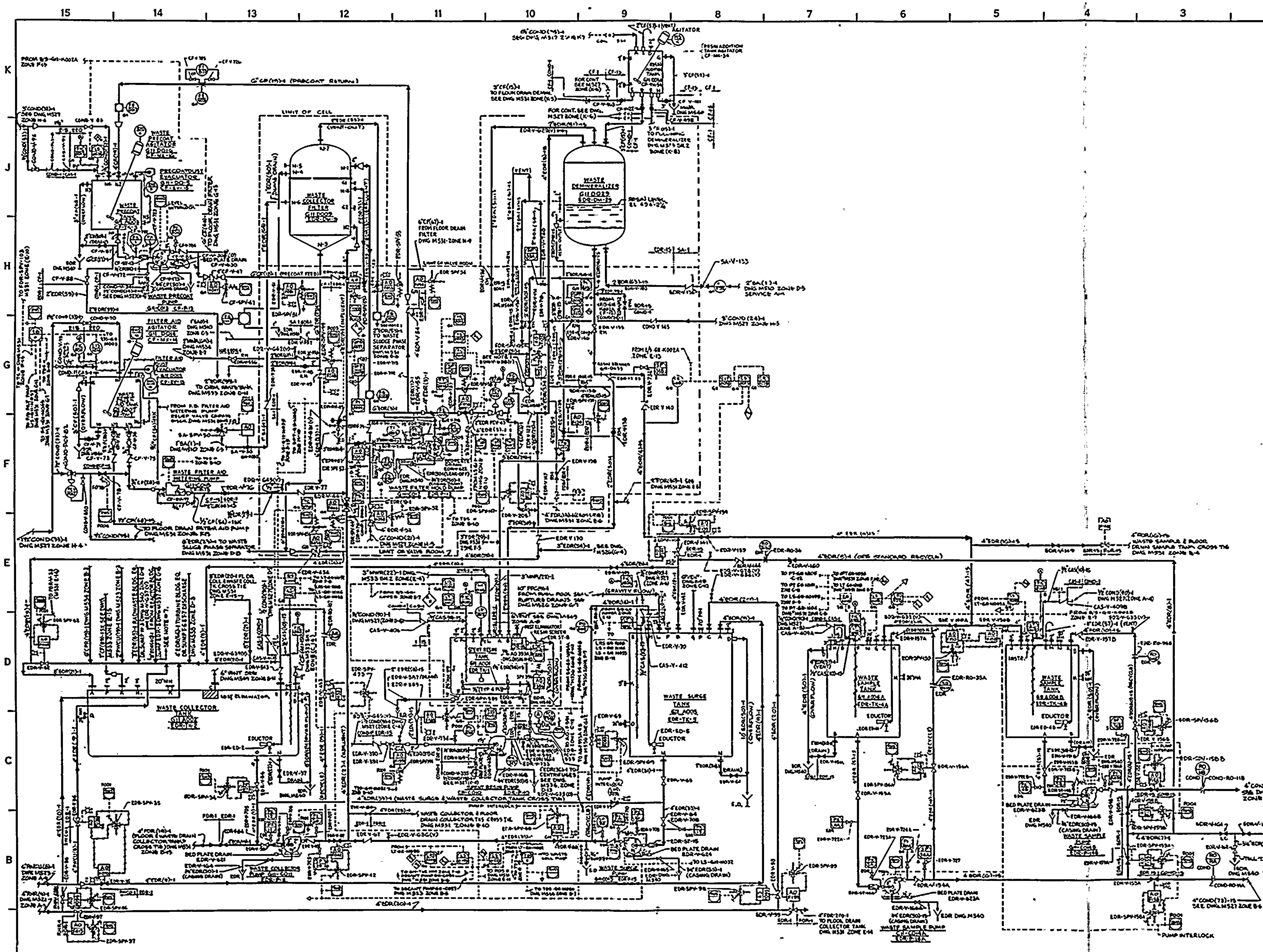
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM HEATER DRAIN SYS. TURBINE GENERATOR BLDG.	FIGURE 10.4-7



2 1

NOTES:

1.- LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA, RETURN PIPING DOWNSTREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.

2- ALL ITEMS MARKED & ARE FURNISHED WITH ASSOCIATED EQUIPMENT

EXCEPT AS NOTED IN NOTE 7.

3 - ALL PIPING, VALVES AND ASSOCIATED COMPONENTS IN THE EDR SYSTEM ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
QUALITY CLASS II
SEISMIC CATEGORY II
CODE GROUP C

4 - ALL PIPING, VALVES & ASSOCIATED COMPONENTS, NOT IN THE EDR SYSTEM ON THIS DWG. SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATIONS SHOWN ON THEIR RESPECTIVE SYSTEM FLOW DIAGRAMS.

5- ALL INSTRUMENT ROOT VALVES NOT LABELED TO BE 3/4" GLOBE VALVES UNLESS OTHERWISE NOTED.

6- ALL INSTRUMENT PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVES SHALL BE CLASSIFIED AS FOLLOWS:
QUALITY CLASS II
SEISMIC CATEGORY II
CODE GROUP D

7- PIPING AND ASSOCIATED COMPONENTS IN SUBSYSTEM EDR(49) SHALL BE CLASSIFIED AS FOLLOWS:
QUALITY CLASS-II
SEISMIC CATEGORY-II
CODE GROUP-D

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A 3/4" VENT VALVE (EXCEPT WHERE NOTED)
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A 3/4" DRAIN VALVE (EXCEPT WHERE NOTED)

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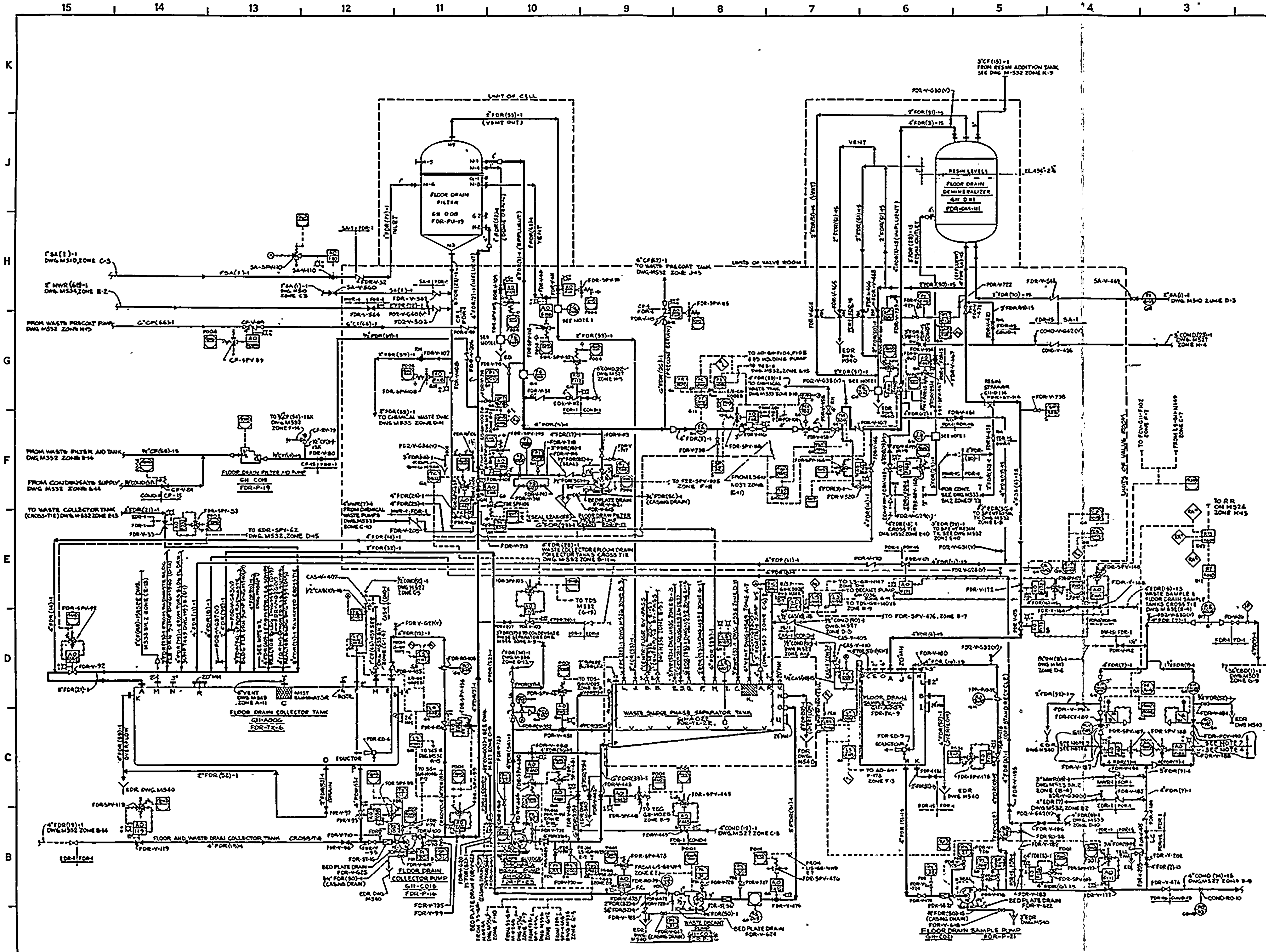
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM RADIOACTIVE WASTE SYSTEM EQUIPMENT DRAIN PROCESSING	FIGURE 11.2-2



NOTES:

1. LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA. RETURN PIPING DOWN STREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.
2. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
3. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE ¾" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
4. EXCEPT AS NOTED IN NOTE 7, ALL PIPING, VALVES AND ASSOCIATED COMPONENTS IN THE FDR SYSTEM ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP C.
5. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS, NOT IN THE FDR SYSTEM ON THIS DWG. SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATIONS SHOWN ON THEIR RESPECTIVE SYSTEM FLOW DIAGRAMS.
6. ALL INSTRUMENT PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVES SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
7. PIPING AND ASSOCIATED COMPONENTS IN SUBSYSTEMS FDR(40) & FDR(41) AND FDR(7) DOWNSTREAM OF FDR-V-187 AND FDR-V-188 SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS-II
   SEISMIC CATEGORY-II
   CODE GROUP-D

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A ¾" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A ¾" DRAIN VALVE.

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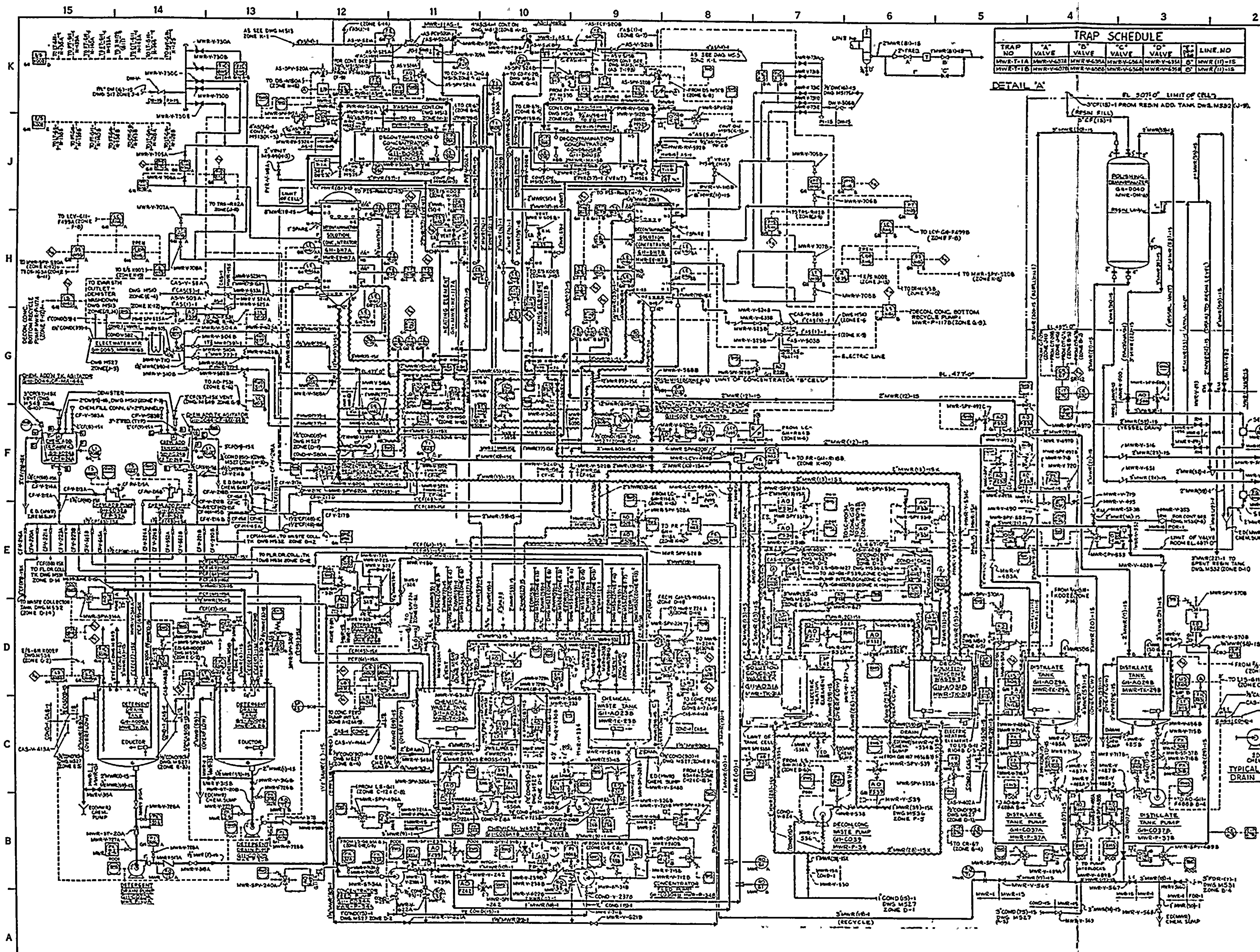
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM RADIOACTIVE WASTE SYSTEM FLOOR DRAIN PROCESSING	FIGURE 11.2-3



TRAP NO	"A" VALVE	"B" VALVE	"C" VALVE	"D" VALVE	"E"	LINE NO
MWR-T-1A	MWR-V-637A	MWR-V-638A	MWR-V-636A	MWR-V-635A	8"	MWR (11)-15
MWR-T-1B	MWR-V-637B	MWR-V-638B	MWR-V-636B	MWR-V-635B	8"	MWR (11)-15

NOTES:

1.- LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA. RETURN PIPING DOWNSTREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.

2. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.

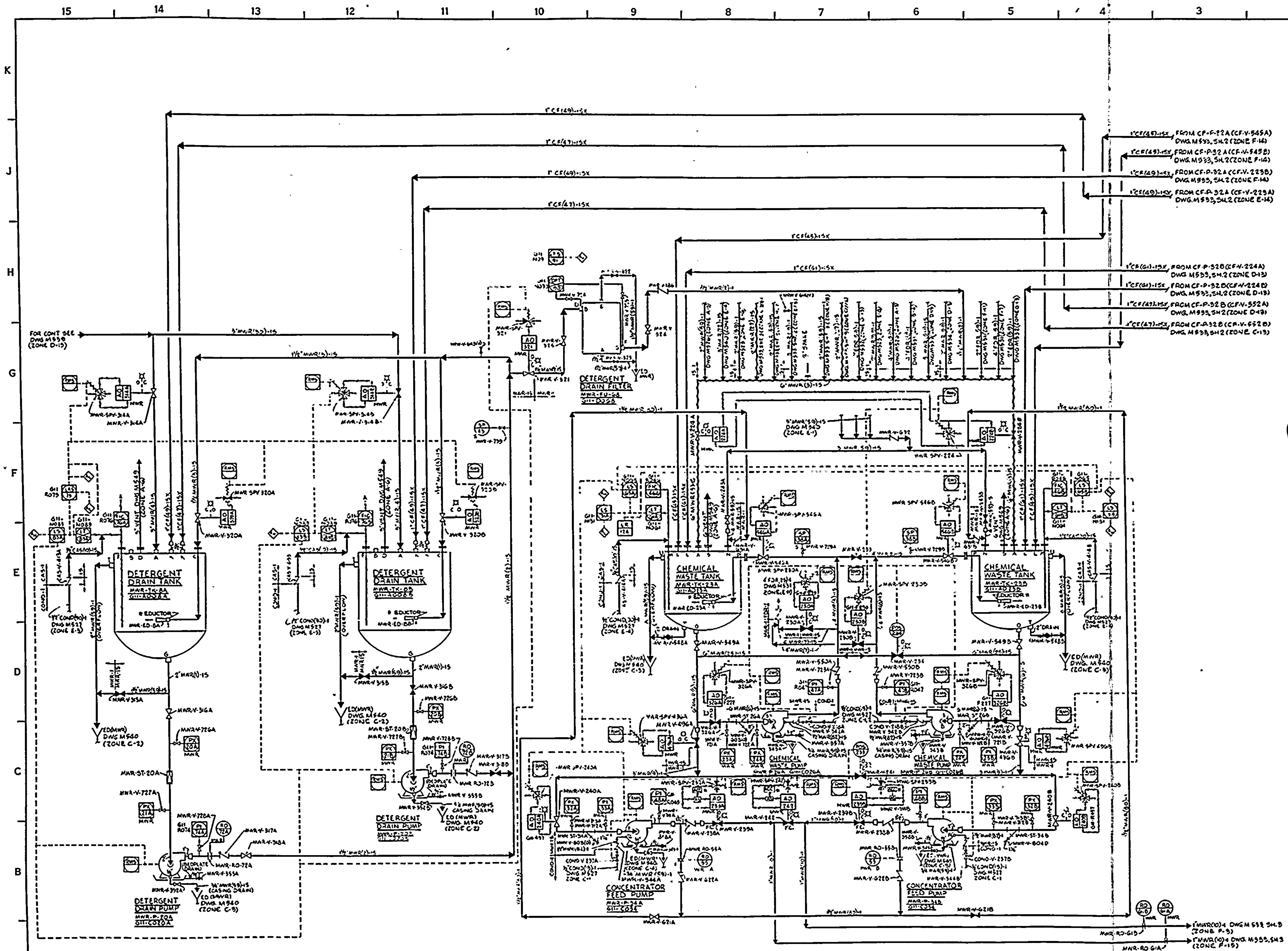
3. ALL PIPING VALVES AND ASSOCIATED COMPONENTS IN THE MWR. AND PVR. SYSTEMS SHOWN ON THIS DWG. ARE CLASSIFIED AS FOLLOWS;
SEISMIC CATEGORY II
QUALITY CLASS II
CODE GROUP C.

4. ALL PIPING VALVES AND ASSOCIATED COMPONENTS NOT IN THE MWR. AND PVR. SYSTEMS ON THIS DWG. SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATIONS SHOWN ON THEIR RESPECTIVE FLOW DIAGRAMS.

5. LOCAL CONDUCTIVITY ELEMENTS, ESPECIALLY IN LINES WITH INTERMITTENT FLOWS, ARE TO BE KEPT WET AT ALL TIMES BY A LOOP SEAL.

6. LINES SHOWN AS ~~~~ ARE ELECTRICAL HEAT TRACED.

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM CHEMICAL WASTE PROCESSING	FIGURE 11.2-4



NOTES:

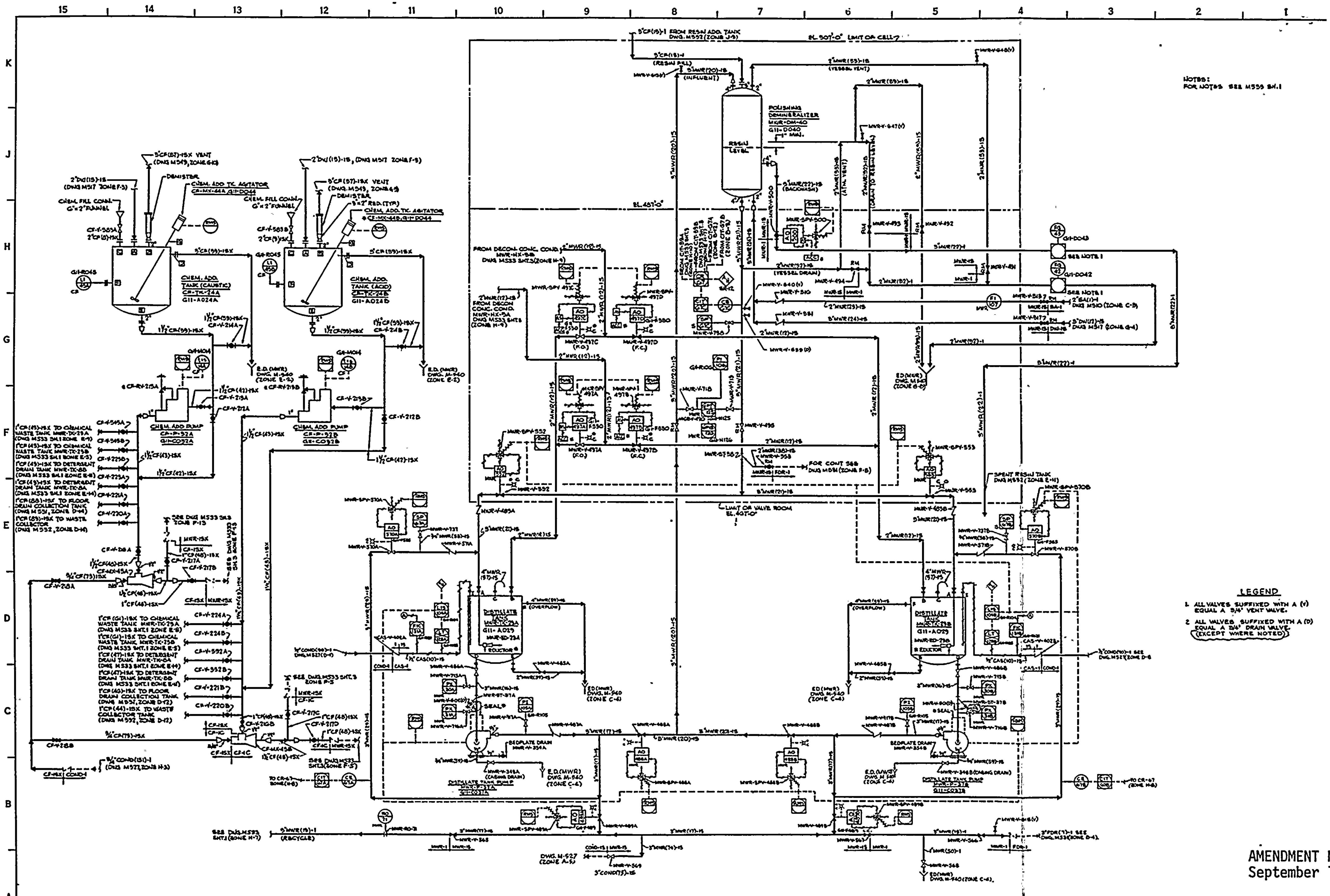
1. LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA. RETURN PIPING DOWNSTREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.
2. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
3. ALL PIPING VALVES AND ASSOCIATED COMPONENTS IN THE MWR AND PVR SYSTEMS SHOWN ON THIS DWG. ARE CLASSIFIED AS FOLLOWS UNLESS OTHERWISE NOTED:
   SEISMIC CATEGORY II
   QUALITY CLASS II
   CODE GROUP C EXCEPT CODE GROUP D FOR INSTRUMENT PIPING AND SUBSYSTEM MWR-(57)-15 AND MWR-(59)-15X PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE CONNECTED TO VESSELS, TANKS, PUMPS, HEAT EXCHANGERS, ETC, AND MWR(30)-15.
4. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS NOT IN THE MWR AND PVR SYSTEMS ON THIS DWG. SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATION SHOWN ON THEIR RESPECTIVE FLOW DIAGRAMS.
5. LOCAL CONDUCTIVITY ELEMENTS, ESPECIALLY IN LINES WITH INTERMITTENT FLOWS, ARE TO BE KEPT WET AT ALL TIMES BY A LOOP SEAL.
6. LINES SHOWN AS ~~~~ ARE ELECTRICAL HEAT TRACED.
7. ALL ITEMS MARKED WITH AN ASTERISK (*) ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
8. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS REFERRING TO THIS NOTE SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY II
   QUALITY CLASS II
   CODE GROUP D
9. ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION "SYSTEM" (50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) EQUAL A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) EQUAL A 3/4" DRAIN VALVE. (EXCEPT WHERE NOTED)

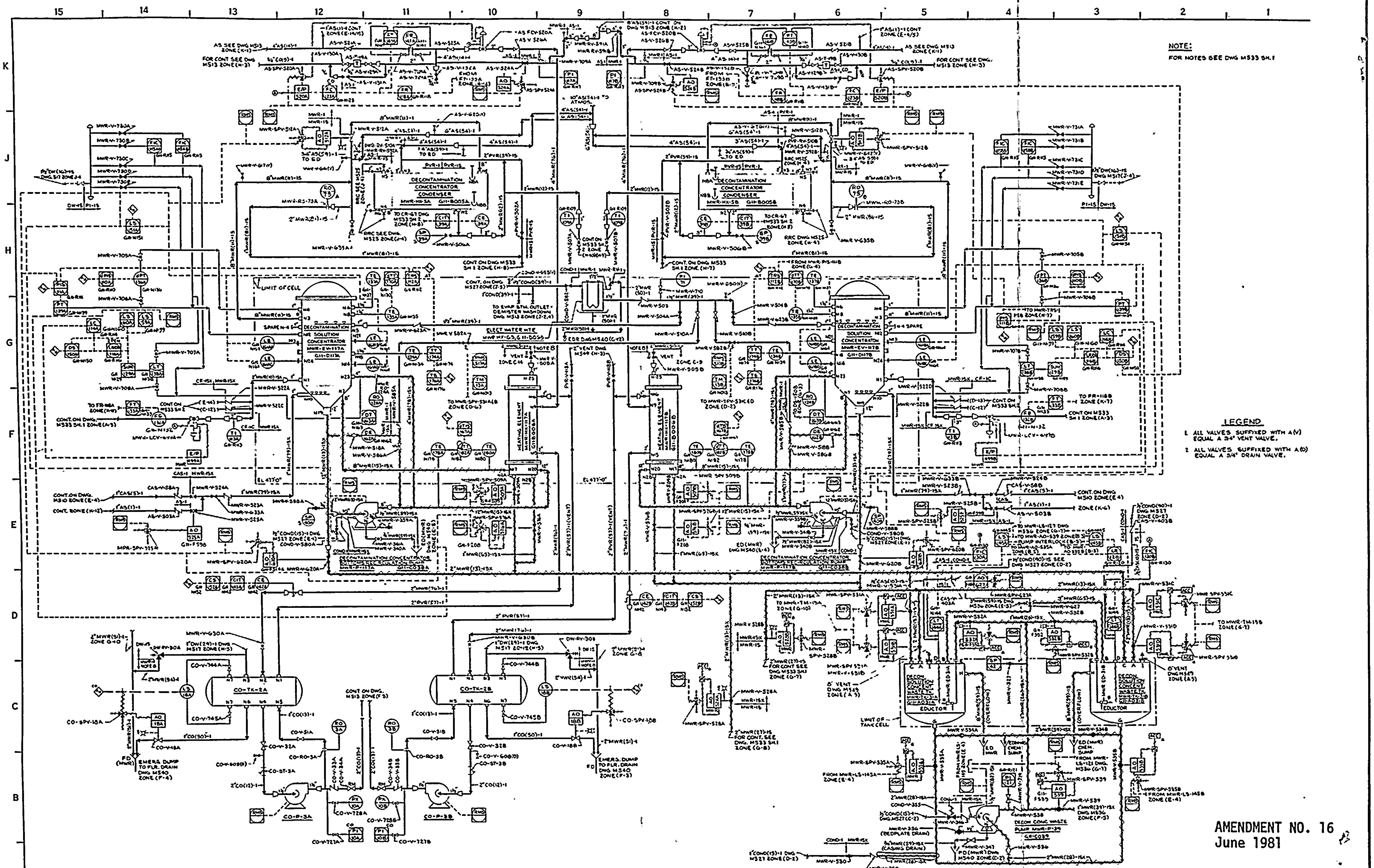
AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM CHEMICAL WASTE PROCESSING	FIGURE 11.2-4a



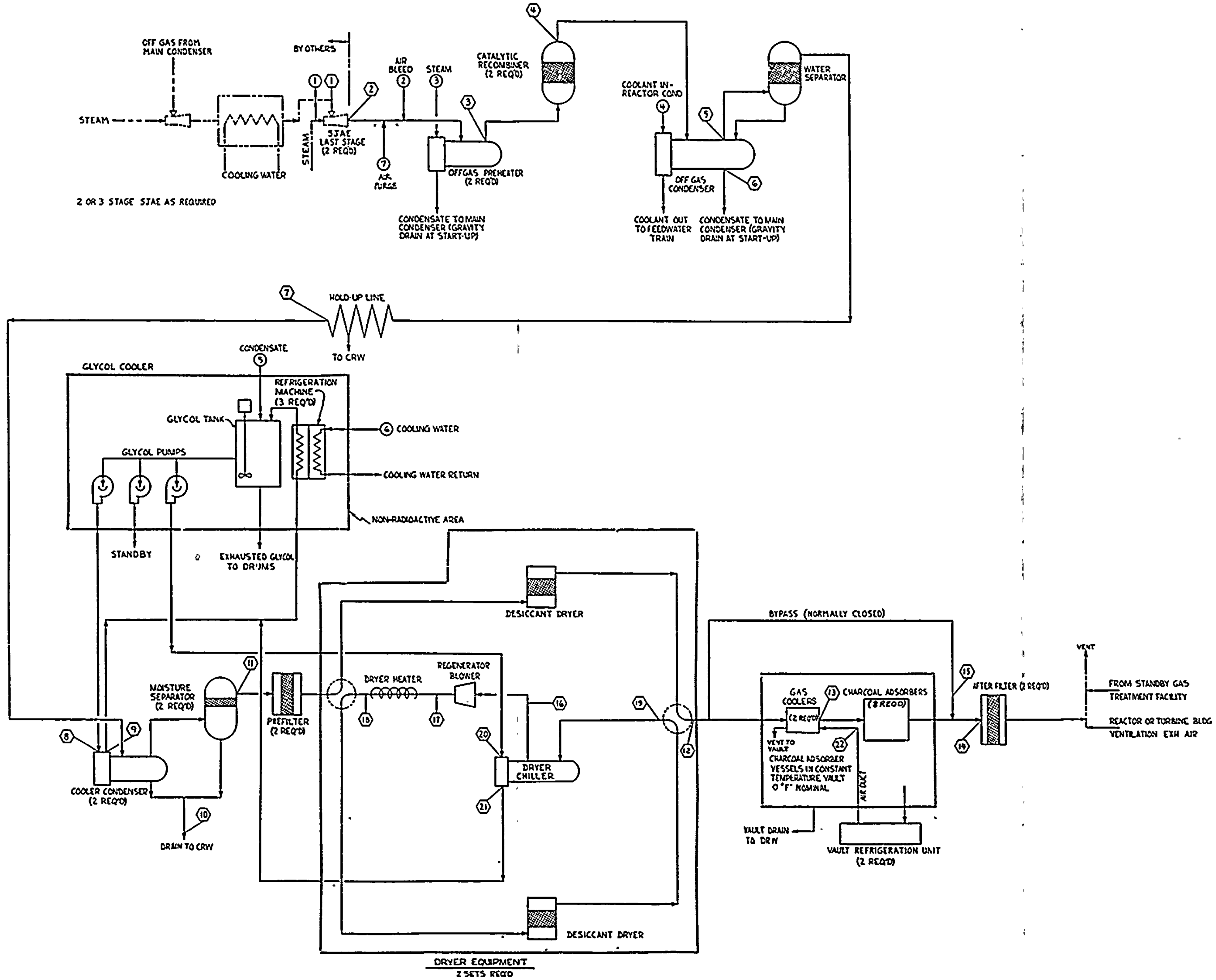
AMENDMENT NO. 11
September 1980

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM CHEMICAL WASTE PROCESSING	FIGURE 11.2-4b

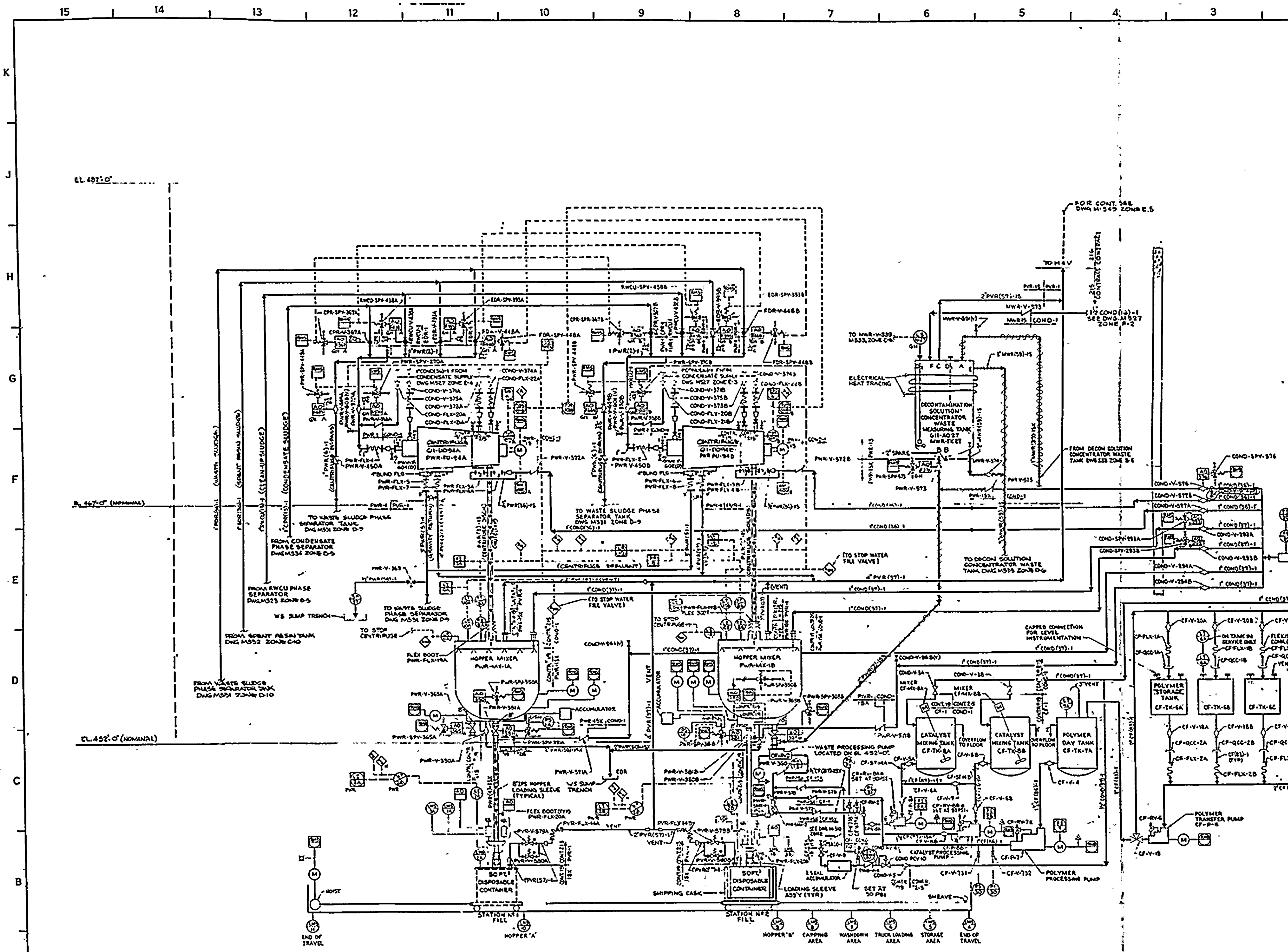


AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW DIAGRAM CHEMICAL WASTE PROCESSING	FIGURE 11.2-4c



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	OFF GAS SYSTEM - LOW TEMPERATURE	FIGURE 11.3-1



NOTES

1. ALL PIPING, VALVES & ASSOCIATED EQUIPMENT SHOWN ON THIS DRAWING EXCEPT AS DELINEATED IN NOTE 4 & 9 ARE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP C
2. ALL PROCESS PIPING DOWNSTREAM OF CENTRIFUGE AND ALL CHEMICAL FEED PIPING (CATALYST & POLYMER) BY CONTRACT #19
3. ALL CONDENSATE PIPING (COND) UP TO CONTRACT BREAK (CONTRACT#19 | CONTRACT#215) BY CONTRACT #215
4. ALL PIPING VALVES AND ASSOCIATED EQUIPMENT IN THE CHEMICAL FEED (CF) AND CONDENSATE (COND) SYSTEMS SHOWN ON THIS DWG ARE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
5. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
6. ALL ITEMS MARKED # ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
7. LINES SHOWN AS ~~~~ ARE ELECTRICAL HEAT TRACED.
8. ALL INSTRUMENTATION PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVE SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATAGORY II
   CODE GROUP D
9. ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION "SYSTEM" (50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING.

LEGEND

1. ALL VALVES SUFFIXED WITH A(V) EQUAL A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A(D) EQUAL A 3/4" DRAIN VALVE.

AMENDMENT NO. 16
June 1981

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	RADIOACTIVE WASTE DISPOSAL SOLIDS HANDLING SYSTEM	FIGURE 11.4-1