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August 28, 1987

U. S. Nuclear Regulatory Commission
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Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1 and 2
Docket Nos. STN 50-528 (License NPF-41)
STN 50-529 (License NPF-51)
Semiannual Radioactive Effluent Release Report
File: 87-E-056-026; 87-F-056-026

Dear Sir:

Pursuant to 10 CFR 50.36(a)(2) and in accordance with Technical Specification 6.9.1.8, attached please find the Semiannual Radioactive Effluent Release Report for the Palo Verde Nuclear Generating Station Units No. 1 and 2 for the six month period ending June 30, 1987.

If you have any questions, please call Mr. W. F. Quinn at (602) 371-4087.

Very truly yours,

JGHaynes

J. G. Haynes
Vice President
Nuclear Production

JGH/PGN/ls
Attachment

cc: O. M. De Michele (all w/a)
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11

PALO VERDE NUCLEAR GENERATING STATION
UNITS 1 AND 2

SEMI-ANNUAL RADIOACTIVE
EFFLUENT RELEASE REPORT
JANUARY 1, 1987 THROUGH JUNE 30, 1987

USNRC Dockets STN-50-528 and STN-50-529

8709080042 870630
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R PDR

Table of Contents

List of Tables

List of Tables (Continued)

INTRODUCTION

This report summarizes meteorological data and doses from radioactive effluents for the Palo Verde Nuclear Generating Station (PVNGS) for the period January through June 1987. The data presented meet the reporting requirements of Regulatory Guide 1.21 of the U.S. Nuclear Regulatory Commission (Revision 1, June 1974) as well as the PVNGS Technical Specifications.

The report is organized into four parts. Appendix A presents the effluent and waste disposal source term data. Appendix B presents a summary of onsite meteorological data for the report period. Appendix C presents the radiological doses from gaseous radioactive effluents. Appendix D presents the revision to the Process Control Program (PCP).

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APPENDIX A

SOURCE TERMS AND

EFFLUENT AND WASTE DISPOSAL REPORTS

Supplemental Information

1.0 Regulatory Limits

1.1 Liquid Releases

a. PVNGS Technical Specification 3.11.1.1

The concentration of radioactive material discharged from secondary system liquid waste to the onsite evaporation ponds shall be limited to the Lower Limit of Detectability (LLD) defined as 5×10^{-7} µCi/ml for the principal gamma emitters or 1×10^{-6} µCi/ml for I-131.

b. PVNGS Technical Specification 3.11.1.2

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY shall be limited:

- During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

- During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

1.2 Gaseous Releases

a. PVNGS Technical Specification 3.11.2.1

The dose rate due to radioactive materials released in gaseous effluents from the site shall be limited to the following:

- For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

- For I-131 and I-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

b. PVNGS Technical Specification 3.11.2.2

The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

- During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

c. PVNGS Technical Specification 3.11.2.3

The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

- During any calendar year: Less than or equal to 15 mrems to any organ.

d. PVNGS Technical Specification 3.11.2.4

The GASEOUS RADWASTE SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, to areas at and beyond the SITE BOUNDARY when averaged over 31 days would exceed 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

1.3 Total Dose

a. PVNGS Technical Specification 3.11.4

The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

2.0 Maximum Permissible Concentrations

Air: Release Concentrations are limited to dose rate limits described in 1.2.a of this report.

3.0 The average energy ($\bar{E}$) of the radionuclide mixture in releases of fission and activation gases is not applicable to PVNGS.

4.0 Measurement and Approximations of Total Radioactivity in Gaseous Effluents.

For continuous releases, sampling is in accordance with PVNGS Technical Specification Table 4.11-2. Particulate and iodine radionuclides are sampled continuously at the three exhaust points. The particulate filters and charcoal cartridges are exchanged for analysis four times per month. Noble gas and tritium are sampled at least once per 31 days. The hourly average Radiation Monitoring System (RMS) effluent monitor readings are used, when available, to account for increases and decreases in noble gas concentrations between noble gas grab samples. The tritium concentration is assumed constant between sampling periods.

For batch releases, sampling is also in accordance with PVNGS Technical Specification Table 4.11-2. For containment purges, the noble gas concentration is adjusted to account for decreases or increases in concentration during the purge using RMS readings. The volume of air released during the purge is determined using the exhaust fan rated flow rate. For Waste Gas Decay Tank releases, the volume released is corrected to standard pressure.

The Lower Limit of Detection (LLD) of a measurement system is defined in Table 4.11-2 of the PVNGS Technical Specifications. An average LLD for each radionuclide is provided in Tables A1 and A5.

5.0 Batch Releases

5.1 Gaseous

	Unit 1*	Unit 2*
o Number of batch releases:	52	56
o Total time period for batch releases:	236,422	105,298
o Maximum time period for a batch release:	10,080	10,080
o Average time period for a batch release:	4,547	1,880
o Minimum time period for a batch release:	5	7

* All times are in minutes

5.2 Liquid

None

6.0 Abnormal Releases

Unit 1 had one abnormal release, from 0845 hours to 0915 hours on April 22, 1987, of Zirconium-95/Niobium-95. The amount released in curies is shown in Table A3.

7.0 Offsite Dose Calculation Manual (ODCM), Process Control Program (PCP) and Preplanned Alternate Sampling Program (PASP) Revisions

There were no revisions to the ODCM or the PASP. There were no major changes to radwaste systems. The only change to the PCP was the utilization of two additional solidification vendors and processes.

These are: U.S. Gypsum, Topical Report 5/84 and Chem Nuclear Systems, CNSI-WF-C-02-P. The PCP changes are in Appendix D.

8.0 Effluents and Solid Wastes

8.1 Gaseous Effluents

The gaseous effluents for the first and second quarters are included in Tables A2, A3, A6, and A7. Included in these tables are summaries of the effluents and estimated total error.

8.2 Liquid Effluents

There were no liquid effluents from the PVNGS site.

8.3 Solid Waste

Solid waste shipments are summarized in Table A10.

9.0 Miscellaneous Information

Releases made to the evaporation pond have been limited at the tank to the concentrations specified in Technical Specification 3.11.1.1. In addition, PVNGS has imposed a limit of 3×10^{-3} μCi/ml for tritium in tanks released to the evaporation pond. This is the maximum permissible concentration for unrestricted areas for tritium in water from 10CFR20 Appendix B. The evaporation pond was monitored in accordance with Technical Specification 3.12.1. During this report period, the analyses showed tritium concentrations in water to be less than 4×10^{-6} μCi/ml.

The results of the fourth quarter 1986 Strontium-89 and Strontium-90 analyses for continuous mode releases were less than the Lower Limit of Detection. The gaseous effluent and dose summaries are therefore correct as reported in the July - December, 1986 Semi-Annual Report.

TABLE A1

PVNGS UNIT 1

GASEOUS EFFLUENTS - LOWER LIMIT OF DETECTION

µCi/cc

NUCLIDE	CONTINUOUS	BATCH
KRYPTON 85	6.17E-06	9.71E-06
KRYPTON 85m	2.83E-08	3.68E-08
KRYPTON 87	6.95E-08	7.68E-08
KRYPTON 88	1.04E-07	1.40E-07
XENON 133	7.02E-08	6.10E-08
XENON 133m	1.84E-07	1.80E-07
XENON 135	2.53E-08	4.29E-08
XENON 135m	5.10E-07	1.65E-06
XENON 138	1.62E-06	1.01E-06
IODINE 131	2.92E-13	1.38E-09
IODINE 133	2.31E-13	4.44E-12
IODINE 135	2.79E-12	6.87E-11
BARIUM 140	2.27E-13	2.90E-11
CERIUM 141	1.09E-13	9.50E-12
CERIUM 144	5.69E-13	4.50E-11
CESIUM 134	1.01E-13	9.20E-12
CESIUM 137	6.48E-14	9.10E-12
COBALT 58	2.57E-13	4.99E-12
COBALT 60	5.81E-13	6.79E-12
IRON 59	6.43E-14	1.70E-11
LANTHANUM 140	1.77E-13	1.40E-11
MANGANESE 54	1.81E-12	5.40E-12
MOLYBDENUM 99	6.39E-13	4.70E-11
STRONTIUM 89	5.00E-16	*
STRONTIUM 90	5.00E-16	*
TRITIUM	3.30E-07	3.97E-06
ZINC 65	1.92E-13	2.20E-11
GROSS ALPHA	7.37E-15	*

*Not required for batch releases.

Table A2
PVNGS UNIT 1
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1987)

GASEOUS EFFLUENTS-SUMMATION OF ALL RELEASES

	Unit	Quarter #1	Quarter #2	Est. Total Error%*
A. Fission & activation gases				
1. Total release	Ci	4.21E+02	1.59E+02	3.97E+01
2. Average release rate for period	µCi/sec	5.41E+01	2.02E+01	
3. Percent of technical specification limit	%	NA**	NA**	
B. Iodines				
1. Total Iodine-131	Ci	3.43E-03	1.24E-02	2.93E+01
2. Average release rate for period	µCi/sec	4.41E-04	1.58E-03	
3. Percent of technical specification limit	%	NA**	NA**	
C. Particulates				
1. Particulates with half-lives >8 days	Ci	2.71E-04	2.79E-04	2.93E+01
2. Average release rate for period	µCi/sec	3.49E-05	3.54E-05	
3. Percent of technical specification limit	%	NA**	NA**	
4. Gross Alpha radioactivity	Ci	<LLD	<LLD	
D. Tritium				
1. Total release	Ci	1.35E+02	7.23E+01	4.22E+01
2. Average release rate for period	µCi/sec	1.74E+01	9.20E+00	
3. Percent of technical specification limit	%	NA**	NA**	

*Estimated total error methodology is presented in Table A9.
**See Table A4 for percent of technical specification limits.

Table A3
PVNGS UNIT 1
GASEOUS EFFLUENTS-GROUND LEVEL RELEASES

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter #1	Quarter #2	Quarter #1	Quarter #2
1. Fission gases					
Argon 41	Ci	3.99E-01	*	2.45E-01	2.52E-01
Krypton-85	Ci	<LLD	<LLD	1.80E+00	3.98E+00
Krypton-85m	Ci	2.35E-01	<LLD	1.21E-01	3.13E-03
Krypton-87	Ci	1.42E-01	<LLD	<LLD	4.02E-03
Krypton-88	Ci	2.87E-01	<LLD	3.83E-03	6.81E-03
Xenon-131m	Ci	*	*	4.05E+01	8.22E-02
Xenon-133	Ci	1.25E+02	1.50E+02	2.41E+02	1.11E+00
Xenon-133m	Ci	4.59E-02	<LLD	2.41E+00	1.32E-03
Xenon-135	Ci	1.81E+00	3.29E+00	6.59E+00	2.04E-02
Xenon-135m	Ci	1.03E-01	<LLD	<LLD	<LLD
Xenon-138	Ci	9.50E-02	<LLD	<LLD	<LLD
Unidentified	Ci	**	**	**	**
Total for period	Ci	1.28E+02	1.53E+02	2.93E+02	5.46E+00
2. Iodines					
Iodine-131	Ci	2.51E-03	1.08E-02	9.20E-04	1.62E-03
Iodine-132	Ci	4.41E-06	*	7.02E-04	1.97E-07
Iodine-133	Ci	4.12E-04	1.58E-03	1.44E-04	4.16E-03
Iodine-135	Ci	<LLD	<LLD	3.60E-06	1.41E-06
Total for period	Ci	2.93E-03	1.24E-02	1.77E-03	5.78E-03

Table A3 (Continued)
PVNGS UNIT 1
GASEOUS EFFLUENTS-GROUND LEVEL RELEASES

Nuclides Released	Unit	CONTINUOUS MODE Quarter #1	CONTINUOUS MODE Quarter #2	BATCH MODE Quarter #1	BATCH MODE Quarter #2
3. Particulates					
Antimony-122	Ci	4.58E-06	*	*	*
Antimony-124	Ci	2.47E-04	6.19E-05	*	7.57E-05
Barium-139	Ci	5.57E-07	*	*	*
Barium-140	Ci	<LLD	<LLD	<LLD	<LLD
Bromine-82	Ci	*	*	5.26E-05	1.58E-05
Cerium-141	Ci	<LLD	<LLD	<LLD	<LLD
Cerium-144	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-134	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-137	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-138	Ci	7.48E-05	*	7.96E-07	1.71E-06
Chromium-51	Ci	*	*	*	3.43E-05
Cobalt-58	Ci	1.93E-05	<LLD	<LLD	3.34E-05
Cobalt-60	Ci	4.93E-06	<LLD	<LLD	6.88E-05
Iron-59	Ci	<LLD	<LLD	<LLD	<LLD
Lanthanum-140	Ci	<LLD	<LLD	<LLD	<LLD
Manganese-54	Ci	1.13E-07	<LLD	<LLD	<LLD
Molybdenum-99	Ci	<LLD	<LLD	<LLD	<LLD
Niobium-95	Ci	*	*	*	1.71E-06
Rubidium-88	Ci	1.40E-04	*	3.02E-04	9.71E-05
Strontium-89	Ci	<LLD	***	**	**
Strontium-90	Ci	<LLD	***	**	**
Tellurium-131m	Ci	3.26E-07	*	*	*

Table A3 (Continued)
PVNGS UNIT 1
GASEOUS EFFLUENTS-GROUND LEVEL RELEASES

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter #1	Quarter #2	Quarter #1	Quarter #2
3. Particulates (Continued)					
Tritium	Ci	<LLD	<LLD	1.35E+02	7.23E+01
Yttrium-90m	Ci	3.16E-06	*	*	*
Zinc-65	Ci	<LLD	<LLD	<LLD	<LLD
Zirconium-95	Ci	*	*	*	2.69E-06
Unidentified	Ci	**	**	**	**
Total for period	Ci	4.96E-04	6.19E-05	1.35E+02	7.23E+01

*Not detected
**Not applicable
***Analysis not yet completed. Additional information will be included in the next Semi-annual Report.

TABLE A4

PVNGS UNIT 1

RADIATION DOSES AT AND BEYOND THE SITE BOUNDARY FOR 1987(1)

	Unit	Quarter #1	(3) Quarter #2	Quarter #3	Quarter #4	(3) Year To Date
1. a. Gamma Air Dose	mrads	4.68E-02	1.76E-02	-	-	6.44E-02
% of T.S. 3.11.2.2	%	9.36E-01	3.52E-01	-	-	6.44E-01
b. Beta Air Dose	mrads	1.31E-01	4.96E-02	-	-	1.81E-01
% of T.S 3.11.2.2	%	1.31E+00	4.96E-01	-	-	9.05E-01
2. Maximum Organ Dose	mrem	(2) 7.69E-01	(2) 2.00E+00	-	-	(2) 2.77E+00
% of T.S. 3.11.2.3	%	1.03E+01	2.67E+01	-	-	1.85E+01

(1) Calculations are based on parameters and methodologies of the ODCM using historical meteorology. The beta and gamma air doses were calculated using the SSW sector X/Q at the site boundary. The maximum organ dose was calculated assuming a residence, garden, milk cow and meat animal using the SSW sector X/Q at the site boundary and the W sector D/Q at the site boundary. These are the highest calculated annual average dispersion parameters for any of the three units. These doses are calculated to estimate percentage of Technical Specification limits.

(2) The limiting organ is the infant's thyroid.

(3) Does not include 2nd quarter Sr-89,90 results.

TABLE A5

PVNGS UNIT 2

GASEOUS EFFLUENTS - LOWER LIMIT OF DETECTION

μCi/cc

NUCLIDE	CONTINUOUS	BATCH
KRYPTON 85	6.27E-06	1.00E-05
KRYPTON 85m	2.66E-08	2.77E-08
KRYPTON 87	5.80E-08	7.60E-06
KRYPTON 88	9.30E-08	2.83E-06
XENON 133	7.25E-08	6.77E-08
XENON 133m	1.80E-07	1.69E-07
XENON 135	2.14E-08	1.98E-08
XENON 135m	3.30E-07	3.20E-06
XENON 138	7.40E-07	8.40E-06
IODINE 131	1.68E-12	2.24E-08
IODINE 133	6.87E-13	1.40E-08
IODINE 135	2.50E-13	5.20E-08
BARIUM 140	1.10E-13	2.10E-09
CERIUM 141	4.10E-14	8.20E-10
CERIUM 144	1.70E-13	3.60E-09
CESIUM 134	5.10E-14	6.80E-10
CESIUM 137	4.50E-14	9.30E-10
COBALT 58	4.80E-14	8.70E-10
COBALT 60	4.80E-14	1.40E-09
IRON 59	9.80E-14	1.80E-09
LANTHANUM 140	7.20E-14	1.80E-09
MANGANESE 54	3.60E-14	9.10E-10
MOLYBDENUM 99	2.60E-13	6.00E-09
STRONTIUM 89	5.00E-16	*
STRONTIUM 90	5.00E-16	*
TRITIUM	4.24E-07	1.40E-06
ZINC 65	9.60E-14	2.00E-09
GROSS ALPHA	7.20E-15	*

*Not required for batch releases.

Table A6
PVNGS UNIT 2
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1987)

GASEOUS EFFLUENTS-SUMMATION OF ALL RELEASES

	Unit	Quarter #1	Quarter #2	Est. Total Error%*
A. Fission & activation gases				
1. Total release	Ci	1.77E+03	1.37E+03	5.88E+01
2. Average release rate for period	µCi/sec	2.27E+02	1.74E+02	
3. Percent of technical specification limit	%	NA**	NA**	
B. Iodines				
1. Total Iodine-131	Ci	4.42E-03	3.91E-03	5.23E+01
2. Average release rate for period	µCi/sec	5.68E-04	4.97E-04	
3. Percent of technical specification limit	%	NA**	NA**	
C. Particulates				
1. Particulates with half-lives >8 days	Ci	1.36E-07	8.69E-06	5.23E+01
2. Average release rate for period	µCi/sec	1.75E-08	1.11E-06	
3. Percent of technical specification limit	%	NA**	NA**	
4. Gross Alpha radioactivity	Ci	<LLD	<LLD	
D. Tritium				
1. Total release	Ci	6.99E+01	7.79E+01	4.22E+01
2. Average release rate for period	µCi/sec	8.99E+00	9.91E+00	
3. Percent of technical specification limit	%	NA**	NA**	

*Estimated total error methodology is presented in Table A9.
**See Table A8 for percent of technical specification limits.

Table A7
PVNGS UNIT 2
GASEOUS EFFLUENTS-GROUND LEVEL RELEASES

Nuclides Released	Unit	CONTINUOUS MODE Quarter #1	CONTINUOUS MODE Quarter #2	BATCH MODE Quarter #1	BATCH MODE Quarter #2
1. Fission gases					
Argon 41	Ci	*	*	1.02E+00	1.73E-01
Krypton-85	Ci	<LLD	<LLD	1.11E+01	7.53E-01
Krypton-85m	Ci	3.28E-01	5.26E+00	8.72E-01	1.28E-02
Krypton-87	Ci	<LLD	5.62E-01	<LLD	<LLD
Krypton-88	Ci	<LLD	4.20E+00	6.58E-02	3.01E-03
Xenon-131m	Ci	*	*	2.01E+01	7.47E-01
Xenon-133	Ci	9.70E+02	1.22E+03	7.31E+02	8.23E+01
Xenon-133m	Ci	1.01E+00	5.12E+00	4.25E+00	6.49E-01
Xenon-135	Ci	1.92E+01	4.76E+01	9.49E+00	3.18E-01
Xenon-135m	Ci	<LLD	2.53E+00	<LLD	<LLD
Xenon-138	Ci	<LLD	<LLD	<LLD	<LLD
Unidentified	Ci	**	**	**	**
Total for period	Ci	9.91E+02	1.29E+03	7.78E+02	8.50E+01
2. Iodines					
Iodine-131	Ci	3.68E-03	3.86E-03	7.38E-04	4.92E-05
Iodine-133	Ci	2.67E-04	1.28E-04	7.61E-05	1.18E-05
Iodine-135	Ci	<LLD	<LLD	<LLD	<LLD
Total for period	Ci	3.95E-03	3.99E-03	8.14E-04	6.10E-05

Table A7 (Continued)
PVNGS UNIT 2
GASEOUS EFFLUENTS-GROUND LEVEL RELEASES

3. Particulates

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter #1	Quarter #2	Quarter #1	Quarter #2
Antimony-124	Ci	2.59E-08	4.61E-06	1.10E-07	*
Barium-140	Ci	<LLD	<LLD	<LLD	<LLD
Bromine-82	Ci	*	*	4.28E-05	1.20E-05
Cerium-141	Ci	<LLD	<LLD	<LLD	<LLD
Cerium-144	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-134	Ci	<LLD	<LLD	<LLD	<LLD
Cesium-137	Ci	<LLD	4.08E-06	<LLD	<LLD
Cesium-138	Ci	*	*	1.93E-06	*
Cobalt-58	Ci	<LLD	<LLD	<LLD	<LLD
Cobalt-60	Ci	<LLD	<LLD	<LLD	<LLD
Iron-59	Ci	<LLD	<LLD	<LLD	<LLD
Lanthanum-140	Ci	<LLD	<LLD	<LLD	<LLD
Manganese-54	Ci	<LLD	<LLD	<LLD	<LLD
Molybdenum-99	Ci	<LLD	<LLD	<LLD	<LLD
Rubidium-88	Ci	*	6.41E-03	3.09E-04	1.31E-03
Strontium-89	Ci	<LLD	***	**	**
Strontium-90	Ci	<LLD	***	**	**
Tritium	Ci	<LLD	<LLD	6.99E+01	7.79E+01
Zinc-65	Ci	<LLD	<LLD	<LLD	<LLD
Unidentified	Ci	**	**	**	**
Total for period	Ci	2.59E-08	6.41E-03	6.99E+01	7.79E+01

*Not detected
**Not applicable
***Analysis not yet completed. Additional information will be included in the next Semi-annual Report.

TABLE A8

PVNGS UNIT 2

RADIATION DOSES AT AND BEYOND THE SITE BOUNDARY FOR 1987(1)

	Unit	Quarter #1	(3) Quarter #2	Quarter #3	Quarter #4	(3) Year To Date
1. a. Gamma Air Dose	mrads	1.90E-01	1.80E-01	-	-	3.70E-01
% of T.S. 3.11.2.2	%	3.80E+00	3.60E+00	-	-	3.70E+00
b. Beta Air Dose	mrads	5.41E-01	4.31E-01	-	-	9.72E-01
% of T.S. 3.11.2.2	%	5.41E+00	4.31E+00	-	-	4.86E+00
2. Maximum Organ Dose	mrem	(2) 7.73E-01	(2) 7.07E-01	-	-	(2) 1.48E+00
% of T.S. 3.11.2.3	%	1.03E+01	9.43E+00	-	-	9.78E+00

(1) Calculations are based on parameters and methodologies of the ODCM using historical meteorology. The beta and gamma air doses were calculated using the SSW sector X/Q at the site boundary. The maximum organ dose was calculated assuming a residence, garden, milk cow and meat animal using the SSW sector X/Q at the site boundary and the W sector D/Q at the site boundary. These are the highest calculated annual average dispersion parameters for any of the three units. These doses are calculated to estimate percentage of Technical Specification limits.

(2) The limiting organ is the infant's thyroid.

(3) Does not include 2nd quarter Sr-89,90 results.

Table A9
Estimation Methodology of Total Percent Error

The estimated total errors is calculated as follows:

$$\text{Total Percent Error} = (E_1^2 + E_2^2 + E_3^2\ ,\ldots,\ + E_n^2)^{1/2}$$

Where E_n = Percent error associated with each contributing parameter.

Parameters contributing to errors in the measurement of gaseous effluents are process flow rates, sample collection, analytical counting and tank volumes.

Table A-10

Solid Waste Summary for January-June 1987

A. SOLID WASTE SHIPPED OFFSITE FOR BURIAL OR DISPOSAL (Not irradiated fuel)

1. Type of Waste	Unit	6-Month Period	Estimated Total Error %
a. Spent resins, filter sludges, evaporator bottoms, etc.	M^3	1.12E+02	
	Ci	4.01E+00*	2.50E+01
b. Dry compressible waste, contaminated equipment, etc.	M^3	0.00E+00	
	Ci	0.00E+00	NA
c. Irradiated components, control rods, etc.	M^3	0.00E+00	
	Ci	0.00E+00	NA
d. Other (filters in a HIC)	M^3	1.41E+00	
	Ci	1.43E+01*	2.50E+01

*As determined by measurement; see A.2 for scaled nuclides.

2. a. Estimate of Major Nuclide Composition for Spent Resins, Filter Sludges, Evaporator Bottoms, etc., as Determined by Measurement

Nuclide	% [a]	Curies
Tritium	99.50	3.99
Sb124	0.26	1.03E-2
(see note)	0.34	1.36E-2
Total		4.01

Note: The following nuclides make up the 0.34% at 1.36E-2 Ci: Pu-239/240=5.96E-8, Pu-238=1.51E-8, Am-241=8.58E-10, Cm-243/244=1.49E-9, Cm-242=2.95E-8, Sr-90=2.86E-6, Co-58=9.42E-4, Co-60=5.80E-4, Cs-137=5.12E-3, Mn-54=2.43E-3, Tc-99=6.16E-10, C-14=1.00-3, I-129=1.28E-9, I-131=5.20E-5, Cr-51=5.92E-5, Fe-59=3.64E-6, Ni-63=2.95E-4, Nb-95=1.46E-5, Zr-95=1.30E-5, Fe-55=3.25E-4, Pu-241=9.70E-7, Zn-65=4.00E-6, Cs-134=2.74E-3, La-140=5.94E-10, Cs-136=2.06E-5, Na-24=1.21E-16, Sb-122=8.27E-9. Ni-63 and Fe-55 are derived using scaling factors in accordance with 10CFR61.55.

[a] There is a 0.1% error from rounding.

Table A-10 (Continued)

Solid Waste Summary for January-June 1987

2. b. Estimate of Major Nuclide Composition for Filters in an HIC, as Determined by Measurement.

Nuclide	% [a]	Curies
Co60	38.36	5.47
**Ni63	21.88	3.12
**Fe55	19.00	2.71
Co58	9.75	1.39
Mn54	4.91	7.00E-1
Sb124	3.21	4.58E-1
Cs137	2.32	3.31E-1
(see note)	0.55	7.80E-2
Total		14.26

Note: The following nuclides make up 0.55% at 7.80E-2 Ci: Pu-239/240=3.93E-5, Pu-238=9.92E-6, Am-241=5.66-7, Cm-243/244=9.61E-7, Cm-242=2.75E-6, Sr-90=1.85E-3, Tc-99=4.07E-7, C-14=1.65E-6, I-131=3.32E-15, Cr-51=3.67E-3, Fe-59=1.96E-2, Nb-95=1.44E-3, Zr-95=5.11E-2, Pu-241=3.21E-4, I-129=8.48E-8.

[a] There is 0.02% error from rounding.

**Isotopic concentration derived using scaling factors in accordance with 10 CFR 61.55.

3. Solid Waste Disposition

Number of Shipments	Mode of Transportation	Destination
8	Exclusive use Truck	Hanford Reservation Richland, Washington (U.S. Ecology)

B. IRRADIATED FUEL SHIPMENTS (Disposition)

Number of Shipments	Mode of Transportation	Destination
0	NA	NA

C. ADDITIONAL INFORMATION

Eight shipments:

1. 87-RW-001, one HIC at 49.9 cubic foot, filters.

Table A-10 (Continued)

Solid Waste Summary for January-June 1987

2. 87-RW-002, two at 202 cubic foot each, steel liners, solidified resin, cement.

3. 87-RW-003, one at 202 and two at 174 cubic foot each, steel liners, dewatered resin.

4. 87-RW-004, three at 202 cubic foot each, steel liners, dewatered resin.

5. 87-RW-005, three at 202 cubic foot each, steel liners, dewatered resin.

6. 87-RW-006, three at 202 cubic foot each, steel liners, dewatered resin.

7. 87-RW-007, one at 174 and two 202 cubic foot each, steel liners, dewatered resin.

8. 97-RW-008, three at 202 cubic foot each, steel liners, dewatered resin.

Table A-11
PVNGS UNITS 1 AND 2
EFFLUENT MONITORING INSTRUMENTATION OUT OF SERVICE GREATER THAN 30 DAYS

TO BE SUPPLIED LATER.

APPENDIX B

METEOROLOGY

JOINT FREQUENCY DISTRIBUTION TABLES

The tables presented in this section are results obtained from processing the hourly meteorological data collected at the Palo Verde Nuclear Generating Station for the period January-June 1987. The joint frequency distribution (JFD) tables represent the frequency, in terms of the number of observations, that a particular wind speed, wind direction, and stability category occurred simultaneously. On a quarterly, semi-annual and annual basis, the JFDs were produced for 35-foot wind speed and wind direction by atmospheric stability class corresponding to the seven Pasquill stability categories, and for wind speed and wind direction for all stability classes combined. Atmospheric stability was classified per Regulatory Guide 1.23, using the 200-foot to 35-foot temperature difference (delta T).

In accordance with NUREG-0133, the batch releases for the first and second quarters for 1987 were considered as "long term", since for each quarter, the sum of the batch release periods exceeded 150 hours. Consequently, the JFDs for the batch releases for both quarters are the same as for the continuous releases.

Table B1

JFDs of 35-Foot Wind Versus Delta T

January - March 1987

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR1 1987
E IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 3/31/87

*** 1ST QTR 1987 ***

STABILITY CLASS A

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3.51- 4.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4.51- 5.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5.51- 6.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6.51- 8.50	0	1	0	0	0	0	0	0	0	0	0	0	0	0	1	0	2
8.51-11.50	0	0	0	2	1	0	0	0	0	2	1	0	0	2	1	0	9
11.51-14.50	3	0	1	0	0	0	0	0	0	2	4	0	4	1	3	1	19
14.51-20.50	2	0	0	0	1	0	0	0	0	3	5	0	1	8	6	8	34
>20.50	0	0	0	0	0	0	0	0	0	0	4	3	0	0	0	0	7
TOTAL	5	1	1	2	2	0	0	0	0	7	14	3	5	11	11	9	71

STABILITY CLASS B

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
- 3.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
- 4.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1- 5.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
5.51- 6.50	0	0	0	0	0	1	0	0	0	0	0	1	0	0	0	2	4
6.51- 8.50	0	1	0	0	2	0	0	0	0	0	0	2	0	0	1	1	7
8.51-11.50	0	1	0	0	2	1	0	0	1	0	0	0	1	0	2	1	9
11.51-14.50	0	1	1	0	1	2	0	0	0	0	4	1	1	0	0	0	11
14.51-20.50	0	1	2	0	4	0	0	0	0	0	2	0	0	1	0	1	11
>20.50	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	1
TOTAL	0	4	3	0	9	4	0	0	1	0	7	4	2	1	3	5	43

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR1 1987
[illegible] IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 3/31/87

*** 1ST QTR 1987 ***

STABILITY CLASS C

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	1	0	0	0	0	0	0	0	0	0	0	1
3.51- 4.50	0	1	0	0	0	0	0	2	0	0	0	0	0	0	0	0	3
4.51- 5.50	1	1	0	0	1	0	1	0	1	3	2	1	0	1	2	0	14
5.51- 6.50	0	0	1	0	0	0	0	0	4	2	1	0	0	0	2	1	11
6.51- 8.50	2	0	1	0	4	5	2	2	1	1	5	5	0	1	1	1	31
8.51-11.50	0	1	0	0	3	2	0	0	0	1	2	0	1	1	1	2	14
11.51-14.50	0	1	1	0	5	0	0	0	0	2	3	1	3	3	0	1	20
14.51-20.50	0	0	1	0	6	1	0	0	0	0	0	2	0	2	1	1	14
>20.50	0	0	0	0	3	0	0	0	0	1	0	0	0	0	0	0	4
TOTAL	3	4	4	0	22	9	3	4	6	10	13	9	4	8	7	6	112

STABILITY CLASS D

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	1
1.51- 2.50	3	1	2	1	3	5	0	3	2	4	1	2	2	0	2	0	31
2.51- 3.50	3	3	6	10	5	11	11	10	9	11	11	3	4	2	2	4	105
3.51- 4.50	4	4	7	9	9	12	10	9	14	15	8	2	3	3	0	3	112
4.51- 5.50	4	1	3	8	11	13	6	6	14	10	4	6	5	6	1	2	100
5.51- 6.50	2	3	1	3	8	4	2	1	4	5	2	3	3	1	1	1	44
6.51- 8.50	1	2	3	2	15	4	2	1	1	4	7	0	2	3	5	3	55
8.51-11.50	1	3	0	2	8	5	2	0	1	3	4	8	5	1	3	2	48
11.51-14.50	1	2	2	2	22	2	2	1	1	3	2	4	1	5	2	0	52
14.51-20.50	0	0	0	0	16	1	0	0	1	2	4	5	1	5	3	2	40
>20.50	0	0	0	0	1	0	0	0	4	2	1	2	0	0	1	0	11
TOTAL	19	19	24	37	98	57	35	31	51	60	44	35	26	26	20	17	599

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR1 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 3/31/87

*** 1ST QTR 1987 ***

STABILITY CLASS E

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1
1.51- 2.50	3	4	1	3	1	2	3	3	0	1	1	2	1	1	4	1	31
2.51- 3.50	2	6	4	3	0	3	4	3	5	0	1	0	0	2	1	1	35
3.51- 4.50	4	3	4	4	1	3	3	1	6	3	0	2	3	2	1	2	42
4.51- 5.50	4	2	3	1	1	2	1	0	0	2	3	2	3	3	1	5	33
5.51- 6.50	2	3	2	2	1	1	3	1	1	2	3	3	1	0	1	1	27
6.51- 8.50	3	3	1	4	0	2	1	0	3	3	14	5	5	4	3	1	52
8.51-11.50	0	1	1	6	5	2	1	1	2	7	17	4	5	2	5	1	60
11.51-14.50	0	0	0	0	3	0	1	0	0	5	4	9	4	3	2	3	34
14.51-20.50	0	0	1	0	5	0	0	0	3	1	1	2	4	8	3	0	28
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	18	23	17	23	17	15	17	9	20	24	44	29	26	25	21	15	343

STABILITY CLASS F

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
76- 1.50	0	0	0	0	0	1	0	0	0	0	0	0	0	1	0	0	2
- 2.50	10	4	0	2	2	1	1	3	3	0	2	5	3	3	3	5	47
- 3.50	14	7	8	3	2	2	7	2	5	5	1	2	4	3	7	9	81
- 4.50	10	3	4	1	4	2	2	0	1	0	3	1	4	3	3	4	45
51- 5.50	8	7	3	2	1	0	1	0	1	1	0	0	3	4	4	6	41
5.51- 6.50	5	3	2	2	1	1	0	0	0	2	4	3	1	2	4	6	36
6.51- 8.50	1	2	3	4	1	0	0	1	0	1	0	1	6	4	3	9	36
8.51-11.50	1	1	1	0	1	0	0	0	0	1	4	2	0	1	2	8	22
11.51-14.50	0	0	0	0	0	0	0	0	0	0	0	0	1	0	0	2	3
14.51-20.50	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	1
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	49	27	22	14	12	7	11	6	10	10	14	14	22	21	26	49	314

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR1 1987
IDENTIFIER: PVNGS
ERIOD EXAMINED: 1/ 1/87 - 3/31/87

*** 1ST QTR 1987 ***

STABILITY CLASS G

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	2	2	2	1	0	0	0	0	1	0	0	0	1	2	3	0	14
1.51- 2.50	18	13	9	3	1	1	4	6	3	3	1	1	6	9	11	23	112
2.51- 3.50	55	45	29	5	4	3	5	4	2	4	2	6	4	3	15	28	214
3.51- 4.50	37	49	29	5	2	1	3	1	2	0	1	4	0	4	12	10	160
4.51- 5.50	27	41	13	2	1	0	0	0	1	0	0	1	1	1	1	5	94
5.51- 6.50	11	12	11	0	0	0	0	0	0	0	0	0	1	0	2	7	44
6.51- 8.50	6	7	7	1	0	0	0	0	0	0	0	0	0	0	1	5	27
8.51-11.50	0	4	0	2	0	0	0	1	0	0	0	0	0	0	0	3	10
11.51-14.50	0	2	0	1	0	0	0	0	0	0	0	0	0	0	0	0	3
14.51-20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	156	175	100	20	8	5	12	12	9	7	4	12	13	19	45	81	678

STABILITY CLASS ALL

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
[illegible]6- 1.50	2	3	2	1	0	1	0	0	1	1	0	0	1	3	3	0	18
[illegible]- 2.50	34	22	12	9	7	9	8	15	8	8	5	10	12	13	20	29	221
[illegible] 3.50	74	61	47	21	11	20	27	19	21	20	15	11	12	10	25	42	436
[illegible] 4.50	55	60	44	19	16	18	18	13	23	18	12	9	10	12	16	19	362
[illegible]1- 5.50	44	52	22	13	15	15	9	6	17	16	9	10	12	15	9	18	282
5.51- 6.50	20	21	17	7	10	7	5	2	9	11	10	10	6	3	10	18	166
6.51- 8.50	13	16	15	11	22	11	5	4	5	9	26	13	13	12	15	20	210
8.51-11.50	2	11	2	12	20	10	3	2	4	14	28	14	12	7	14	17	172
11.51-14.50	4	6	5	3	31	4	3	1	1	12	17	15	14	12	7	7	142
14.51-20.50	2	1	5	0	32	2	0	0	4	6	12	9	6	24	13	12	128
>20.50	0	0	0	0	4	0	0	0	4	3	6	5	0	0	1	0	23
TOTAL	250	253	171	96	168	97	78	62	97	118	140	106	98	111	133	182	2160

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR1 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 3/31/87

*** 1ST QTR 1987 ***

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

TOTAL NUMBER OF OBSERVATIONS: 2160

TOTAL NUMBER OF VALID OBSERVATIONS: 2160

TOTAL NUMBER OF MISSING OBSERVATIONS: 0

PERCENT DATA RECOVERY FOR THIS PERIOD: 100.0 %

MEAN WIND SPEED FOR THIS PERIOD: 6.2 MPH

TOTAL NUMBER OF OBSERVATIONS WITH BACKUP DATA: 0

PERCENTAGE OCCURRENCE OF STABILITY CLASSES

A	B	C	D	E	F	G
3.29	1.99	5.19	27.73	15.88	14.54	31.39

DISTRIBUTION OF WIND DIRECTION VS STABILITY

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
A	5	1	1	2	2	0	0	0	0	7	14	3	5	11	11	9	0
B	0	4	3	0	9	4	0	0	1	0	7	4	2	1	3	5	0
C	3	4	4	0	22	9	3	4	6	10	13	9	4	8	7	6	0
D	19	19	24	37	98	57	35	31	51	60	44	35	26	26	20	17	0
E	18	23	17	23	17	15	17	9	20	24	44	29	26	25	21	15	0
F	49	27	22	14	12	7	11	6	10	10	14	14	22	21	26	49	0
G	156	175	100	20	8	5	12	12	9	7	4	12	13	19	45	81	0
TOTAL	250	253	171	96	168	97	78	62	97	118	140	106	98	111	133	182	0

Table B2

JFDs of 35-Foot Wind Versus Delta T

April - June 1987

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR2 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 4/ 1/87 - 6/30/87

*** 2ND QTR 1987 ***

STABILITY CLASS A

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1
3.51- 4.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4.51- 5.50	1	0	0	0	0	1	1	0	1	0	0	0	0	0	0	0	4
5.51- 6.50	0	0	0	1	1	0	0	0	0	0	2	1	0	0	0	0	5
6.51- 8.50	0	1	0	0	2	2	1	1	2	6	5	4	1	0	0	0	25
8.51-11.50	0	1	0	2	7	4	2	0	2	20	27	12	1	1	0	0	79
11.51-14.50	0	0	0	1	4	4	0	1	3	5	18	3	1	0	0	0	40
14.51-20.50	0	0	0	2	9	1	1	0	0	4	15	1	0	2	1	0	36
>20.50	0	0	0	0	0	0	0	0	0	0	2	1	0	0	0	0	3
TOTAL	1	2	0	6	23	12	5	2	8	35	69	22	3	3	2	0	193

STABILITY CLASS B

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
3.51- 4.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1
4.51- 5.50	1	0	0	2	0	0	3	2	0	1	0	1	0	0	0	0	10
5.51- 6.50	1	0	0	2	1	4	1	4	5	3	5	2	0	0	0	1	29
6.51- 8.50	1	0	3	1	2	4	2	7	19	17	20	7	6	1	1	1	92
8.51-11.50	0	0	1	2	7	6	4	3	6	17	15	6	1	0	0	0	68
11.51-14.50	0	0	1	1	9	3	0	0	0	1	6	2	1	0	0	0	24
14.51-20.50	0	0	0	1	5	3	0	0	0	1	3	2	1	0	0	0	16
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	3	0	5	9	24	20	10	16	30	40	49	20	9	1	2	2	240

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR2 1987
IDENTIFIER: PVNGS
ERIOD EXAMINED: 4/ 1/87 - 6/30/87

*** 2ND QTR 1987 ***

STABILITY CLASS C

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	0	0	2	0	0	0	0	0	0	0	0	2
3.51- 4.50	0	1	1	0	1	0	0	0	1	2	1	0	0	0	0	0	7
4.51- 5.50	1	1	1	2	0	1	3	9	15	12	3	1	0	2	0	1	52
5.51- 6.50	1	1	2	1	1	2	4	11	11	13	8	9	1	0	0	0	65
6.51- 8.50	0	0	0	2	2	6	14	4	8	12	11	4	3	0	0	0	66
8.51-11.50	0	0	2	0	5	0	3	2	0	3	7	6	2	1	1	0	32
11.51-14.50	0	0	0	1	2	1	0	1	0	1	2	0	0	0	0	0	8
14.51-20.50	0	0	0	0	6	0	0	0	0	0	0	0	0	0	1	0	7
>20.50	0	0	0	0	2	0	0	0	0	0	0	0	1	0	0	0	3
TOTAL	2	3	6	6	19	10	24	29	35	43	32	20	7	3	2	1	242

STABILITY CLASS D

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
76- 1.50	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1
- 2.50	2	1	1	0	0	0	1	1	0	2	3	1	1	2	0	0	15
- 3.50	4	3	2	5	2	2	2	3	7	7	4	1	2	1	0	3	48
- 4.50	3	2	2	3	1	4	3	13	9	8	4	5	3	4	2	1	67
1- 5.50	2	0	1	2	2	0	3	8	15	8	3	4	2	2	1	1	54
5.51- 6.50	0	1	3	1	0	0	2	7	5	3	6	0	4	1	1	1	35
6.51- 8.50	1	2	3	4	3	0	4	5	0	2	6	7	2	2	0	0	41
8.51-11.50	0	1	1	0	2	3	3	3	1	0	15	10	4	0	2	0	45
11.51-14.50	0	1	0	7	9	2	1	1	0	3	10	5	1	0	0	1	41
14.51-20.50	0	1	2	4	16	0	1	0	3	4	12	6	4	0	0	1	54
>20.50	0	0	0	0	2	0	0	0	0	0	1	0	0	0	0	0	3
TOTAL	13	12	15	26	37	11	20	41	40	37	64	39	23	12	6	8	404

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR2 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 4/ 1/87 - 6/30/87

*** 2ND QTR 1987 ***

STABILITY CLASS E

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	1
1.51- 2.50	1	2	2	0	0	0	2	2	2	0	1	1	1	1	1	2	18
2.51- 3.50	0	1	2	1	1	0	2	0	1	3	5	3	4	2	2	2	29
3.51- 4.50	1	2	4	1	0	1	1	1	2	2	6	2	1	1	1	2	28
4.51- 5.50	0	2	3	1	0	0	5	3	0	3	5	3	3	0	1	3	32
5.51- 6.50	2	1	1	1	0	0	1	0	2	9	7	5	5	1	0	0	35
6.51- 8.50	1	2	4	1	2	2	2	1	3	6	18	17	8	1	1	2	71
8.51-11.50	3	3	2	10	1	2	3	3	5	11	35	36	9	1	2	2	128
11.51-14.50	0	2	2	12	8	2	1	1	0	11	15	16	2	2	0	1	75
14.51-20.50	0	0	0	1	8	0	0	0	0	5	6	2	0	0	0	0	22
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	8	15	20	29	20	7	17	11	15	50	98	85	33	9	8	14	439

STABILITY CLASS F

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	1	0	1	0	0	0	0	0	0	0	0	0	0	0	1	3
1.51- 2.50	5	3	2	0	3	0	2	3	1	1	1	0	5	3	3	4	36
2.51- 3.50	7	4	4	2	1	1	0	0	0	4	3	7	3	4	7	7	54
3.51- 4.50	8	2	1	1	0	0	0	0	2	6	5	5	8	7	4	7	56
4.51- 5.50	4	2	1	1	1	1	0	1	3	6	11	9	6	3	2	6	57
5.51- 6.50	2	2	2	0	0	0	0	2	0	2	5	7	9	1	0	4	36
6.51- 8.50	2	4	1	0	0	0	0	1	0	7	24	7	12	4	0	2	64
8.51-11.50	4	2	2	0	0	0	0	1	0	3	7	7	4	0	0	3	33
11.51-14.50	1	1	1	1	0	1	0	0	0	1	0	0	0	0	0	0	6
14.51-20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	33	21	14	6	5	3	2	8	6	30	56	42	47	22	16	34	345

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR2 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 4/ 1/87 - 6/30/87

*** 2ND QTR 1987 ***

STABILITY CLASS G

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	2	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	4
1.51- 2.50	7	1	4	1	2	0	0	0	1	0	1	3	6	4	4	8	42
2.51- 3.50	23	10	6	2	1	1	1	1	1	1	3	2	7	7	14	15	95
3.51- 4.50	20	10	5	1	0	1	0	1	3	0	1	3	2	6	14	11	78
4.51- 5.50	21	10	5	1	0	1	0	0	2	0	0	1	2	1	2	2	48
5.51- 6.50	8	12	1	0	0	0	0	0	0	0	0	2	0	0	0	4	27
6.51- 8.50	3	5	6	0	0	0	0	0	0	1	1	0	0	0	0	1	17
8.51-11.50	0	2	2	0	0	0	1	0	0	0	4	0	0	0	0	0	9
11.51-14.50	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	1
14.51-20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	84	50	30	5	3	3	2	2	7	2	10	11	17	18	35	42	321

STABILITY CLASS ALL

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
6- 1.50	3	1	0	2	0	0	0	0	0	0	0	0	0	0	1	2	9
2.50	15	7	9	1	5	0	5	6	4	3	6	5	13	10	8	14	111
3.50	34	18	14	10	5	4	5	6	9	15	15	13	16	14	24	27	229
4.50	32	17	13	6	2	6	4	15	17	18	17	15	14	18	22	21	237
1- 5.50	30	15	11	9	3	4	15	23	36	30	22	19	13	8	6	13	257
5.51- 6.50	14	17	9	6	3	6	8	24	23	30	33	26	19	3	1	10	232
6.51- 8.50	8	14	17	8	11	14	23	19	32	51	85	46	32	8	2	6	376
8.51-11.50	7	9	10	14	22	15	16	12	14	54	110	77	21	3	5	5	394
11.51-14.50	1	4	5	23	32	13	2	4	3	22	51	26	5	2	0	2	195
14.51-20.50	0	1	2	8	44	4	2	0	3	14	36	11	5	2	2	1	135
>20.50	0	0	0	0	4	0	0	0	0	0	3	1	1	0	0	0	9
TOTAL	144	103	90	87	131	66	80	109	141	237	378	239	139	68	71	101	2184

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR QTR2 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 4/ 1/87 - 6/30/87

*** 2ND QTR 1987 ***

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

TOTAL NUMBER OF OBSERVATIONS: 2184

TOTAL NUMBER OF VALID OBSERVATIONS: 2184

TOTAL NUMBER OF MISSING OBSERVATIONS: 0

PERCENT DATA RECOVERY FOR THIS PERIOD: 100.0 %

MEAN WIND SPEED FOR THIS PERIOD: 7.5 MPH

TOTAL NUMBER OF OBSERVATIONS WITH BACKUP DATA: 0

PERCENTAGE OCCURRENCE OF STABILITY CLASSES

A	B	C	D	E	F	G
8.84	10.99	11.08	18.50	20.10	15.80	14.70

DISTRIBUTION OF WIND DIRECTION VS STABILITY

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
A	1	2	0	6	23	12	5	2	8	35	69	22	3	3	2	0	0
B	3	0	5	9	24	20	10	16	30	40	49	20	9	1	2	2	0
C	2	3	6	6	19	10	24	29	35	43	32	20	7	3	2	1	0
D	13	12	15	26	37	11	20	41	40	37	64	39	23	12	6	8	0
E	8	15	20	29	20	7	17	11	15	50	98	85	33	9	8	14	0
F	33	21	14	6	5	3	2	8	6	30	56	42	47	22	16	34	0
G	84	50	30	5	3	3	2	2	7	2	10	11	17	18	35	42	0
TOTAL	144	103	90	87	131	66	80	109	141	237	378	239	139	68	71	101	0

Table B3

JFDs of 35-Foot Wind Versus Delta T

January - June 1987

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR JAN-JUN 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 6/30/87

*** SEMI-ANNUAL ***

STABILITY CLASS A

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1
3.51- 4.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
4.51- 5.50	1	0	0	0	0	1	1	0	1	0	0	0	0	0	0	0	4
5.51- 6.50	0	0	0	1	1	0	0	0	0	0	2	1	0	0	0	0	5
6.51- 8.50	0	2	0	0	2	2	1	1	2	6	5	4	1	0	1	0	27
8.51-11.50	0	1	0	4	8	4	2	0	2	22	28	12	1	3	1	0	88
11.51-14.50	3	0	1	1	4	4	0	1	3	7	22	3	5	1	3	1	59
14.51-20.50	2	0	0	2	10	1	1	0	0	7	20	1	1	10	7	8	70
>20.50	0	0	0	0	0	0	0	0	0	0	6	4	0	0	0	0	10
TOTAL	6	3	1	8	25	12	5	2	8	42	83	25	8	14	13	9	264

STABILITY CLASS B

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
- 3.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
- 4.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	0	1
51- 5.50	1	0	0	2	0	0	3	2	0	1	0	1	0	0	0	0	10
5.51- 6.50	1	0	0	2	1	5	1	4	5	3	5	3	0	0	0	3	33
6.51- 8.50	1	1	3	1	4	4	2	7	19	17	20	9	6	1	2	2	99
8.51-11.50	0	1	1	2	9	7	4	3	7	17	15	6	2	0	2	1	77
11.51-14.50	0	1	2	1	10	5	0	0	0	1	10	3	2	0	0	0	35
14.51-20.50	0	1	2	1	9	3	0	0	0	1	5	2	1	1	0	1	27
>20.50	0	0	0	0	0	0	0	0	0	0	1	0	0	0	0	0	1
TOTAL	3	4	8	9	33	24	10	16	31	40	56	24	11	2	5	7	283

S JFD: 35FT WIND VS DELTA T (200'-35') FOR JAN-JUN 1987
IDENTIFIER: PVNGS
ERIOD EXAMINED: 1/ 1/87 - 6/30/87

*** SEMI-ANNUAL ***

STABILITY CLASS C

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
1.51- 2.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.51- 3.50	0	0	0	0	0	1	0	2	0	0	0	0	0	0	0	0	3
3.51- 4.50	0	2	1	0	1	0	0	2	1	2	1	0	0	0	0	0	10
4.51- 5.50	2	2	1	2	1	1	4	9	16	15	5	2	0	3	2	1	66
5.51- 6.50	1	1	3	1	1	2	4	11	15	15	9	9	1	0	2	1	76
6.51- 8.50	2	0	1	2	6	11	16	6	9	13	16	9	3	1	1	1	97
8.51-11.50	0	1	2	0	8	2	3	2	0	4	9	6	3	2	2	2	46
11.51-14.50	0	1	1	1	7	1	0	1	0	3	5	1	3	3	0	1	28
14.51-20.50	0	0	1	0	12	1	0	0	0	0	0	2	0	2	2	1	21
>20.50	0	0	0	0	5	0	0	0	0	1	0	0	1	0	0	0	7
TOTAL	5	7	10	6	41	19	27	33	41	53	45	29	11	11	9	7	354

STABILITY CLASS D

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
- 1.50	1	0	0	0	0	0	0	0	0	1	0	0	0	0	0	0	2
- 2.50	5	2	3	1	3	5	1	4	2	6	4	3	3	2	2	0	46
- 3.50	7	6	8	15	7	13	13	13	16	18	15	4	6	3	2	7	153
1- 4.50	7	6	9	12	10	16	13	22	23	23	12	7	6	7	2	4	179
4.51- 5.50	6	1	4	10	13	13	9	14	29	18	7	10	7	8	2	3	154
5.51- 6.50	2	4	4	4	8	4	4	8	9	8	8	3	7	2	2	2	79
6.51- 8.50	2	4	6	6	18	4	6	6	1	6	13	7	4	5	5	3	96
8.51-11.50	1	4	1	2	10	8	5	3	2	3	19	18	9	1	5	2	93
11.51-14.50	1	3	2	9	31	4	3	2	1	6	12	9	2	5	2	1	93
14.51-20.50	0	1	2	4	32	1	1	0	4	6	16	11	5	5	3	3	94
>20.50	0	0	0	0	3	0	0	0	4	2	2	2	0	0	1	0	14
TOTAL	32	31	39	63	135	68	55	72	91	97	108	74	49	38	26	25	1003

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR JAN-JUN 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 6/30/87

*** SEMI-ANNUAL ***

STABILITY CLASS E

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	0	1	0	1	0	0	0	0	0	0	0	0	0	0	0	0	2
1.51- 2.50	4	6	3	3	1	2	5	5	2	1	2	3	2	2	5	3	49
2.51- 3.50	2	7	6	4	1	3	6	3	6	3	6	3	4	4	3	3	64
3.51- 4.50	5	5	8	5	1	4	4	2	8	5	6	4	4	3	2	4	70
4.51- 5.50	4	4	6	2	1	2	6	3	0	5	8	5	6	3	2	8	65
5.51- 6.50	4	4	3	3	1	1	4	1	3	11	10	8	6	1	1	1	62
6.51- 8.50	4	5	5	5	2	4	3	1	6	9	32	22	13	5	4	3	123
8.51-11.50	3	4	3	16	6	4	4	4	7	18	52	40	14	3	7	3	188
11.51-14.50	0	2	2	12	11	2	2	1	0	16	19	25	6	5	2	4	109
14.51-20.50	0	0	1	1	13	0	0	0	3	6	7	4	4	8	3	0	50
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	26	38	37	52	37	22	34	20	35	74	142	114	59	34	29	29	782

STABILITY CLASS F

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
6- 1.50	0	1	0	1	0	1	0	0	0	0	0	0	0	1	0	1	5
- 2.50	15	7	2	2	5	1	3	6	4	1	3	5	8	6	6	9	83
- 3.50	21	11	12	5	3	3	7	2	5	9	4	9	7	7	14	16	135
1- 4.50	18	5	5	2	4	2	2	0	3	6	8	6	12	10	7	11	101
.51- 5.50	12	9	4	3	2	1	1	1	4	7	11	9	9	7	6	12	98
5.51- 6.50	7	5	4	2	1	1	0	2	0	4	9	10	10	3	4	10	72
6.51- 8.50	3	6	4	4	1	0	0	2	0	8	24	8	18	8	3	11	100
8.51-11.50	5	3	3	0	1	0	0	1	0	4	11	9	4	1	2	11	55
11.51-14.50	1	1	1	1	0	1	0	0	0	1	0	0	1	0	0	2	9
14.51-20.50	0	0	1	0	0	0	0	0	0	0	0	0	0	0	0	0	1
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	82	48	36	20	17	10	13	14	16	40	70	56	69	43	42	83	659

[illegible]S JFD: 35FT WIND VS DELTA T (200'-35') FOR JAN-JUN 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 6/30/87

*** SEMI-ANNUAL ***

STABILITY CLASS C

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
0.76- 1.50	4	2	2	1	0	0	0	0	1	0	0	0	1	2	4	1	18
1.51- 2.50	25	14	13	4	3	1	4	6	4	3	2	4	12	13	15	31	154
2.51- 3.50	78	55	35	7	5	4	6	5	3	5	5	8	11	10	29	43	309
3.51- 4.50	57	59	34	6	2	2	3	2	5	0	2	7	2	10	26	21	238
4.51- 5.50	48	51	18	3	1	1	0	0	3	0	0	2	3	2	3	7	142
5.51- 6.50	19	24	12	0	0	0	0	0	0	0	0	2	1	0	2	11	71
6.51- 8.50	9	12	13	1	0	0	0	0	0	1	1	0	0	0	1	6	44
8.51-11.50	0	6	2	2	0	0	1	1	0	0	4	0	0	0	0	3	19
11.51-14.50	0	2	1	1	0	0	0	0	0	0	0	0	0	0	0	0	4
14.51-20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
>20.50	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
TOTAL	240	225	130	25	11	8	14	14	16	9	14	23	30	37	80	123	999

STABILITY CLASS ALL

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

JOINT FREQUENCY DISTRIBUTION OF WIND SPEED AND DIRECTION IN HOURS AT 35.00 FEET

SPEED (MPH)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	TOTAL
CALM																	0
[illegible]- 1.50	5	4	2	3	0	1	0	0	1	1	0	0	1	3	4	2	27
[illegible]- 2.50	49	29	21	10	12	9	13	21	12	11	11	15	25	23	28	43	332
[illegible]- 3.50	108	79	61	31	16	24	32	25	30	35	30	24	28	24	49	69	665
[illegible]- 4.50	87	77	57	25	18	24	22	28	40	36	29	24	24	30	38	40	599
4.51- 5.50	74	67	33	22	18	19	24	29	53	46	31	29	25	23	15	31	539
5.51- 6.50	34	38	26	13	13	13	13	26	32	41	43	36	25	6	11	28	398
6.51- 8.50	21	30	32	19	33	25	28	23	37	60	111	59	45	20	17	26	586
8.51-11.50	9	20	12	26	42	25	19	14	18	68	138	91	33	10	19	22	566
11.51-14.50	5	10	10	26	63	17	5	5	4	34	68	41	19	14	7	9	337
14.51-20.50	2	2	7	8	76	6	2	0	7	20	48	20	11	26	15	13	263
>20.50	0	0	0	0	8	0	0	0	4	3	9	6	1	0	1	0	32
TOTAL	394	356	261	183	299	163	158	171	238	355	518	345	237	179	204	283	4344

PVNGS JFD: 35FT WIND VS DELTA T (200'-35') FOR JAN-JUN 1987
IDENTIFIER: PVNGS
PERIOD EXAMINED: 1/ 1/87 - 6/30/87

*** SEMI-ANNUAL ***

STABILITY BASED ON: DELTA T BETWEEN 200.0 AND 35.0 FEET
WIND MEASURED AT: 35.0 FEET
WIND THRESHOLD AT: 0.75 MPH

TOTAL NUMBER OF OBSERVATIONS: 4344

TOTAL NUMBER OF VALID OBSERVATIONS: 4344

TOTAL NUMBER OF MISSING OBSERVATIONS: 0

PERCENT DATA RECOVERY FOR THIS PERIOD: 100.0 %

MEAN WIND SPEED FOR THIS PERIOD: 6.9 MPH

TOTAL NUMBER OF OBSERVATIONS WITH BACKUP DATA: 0

PERCENTAGE OCCURRENCE OF STABILITY CLASSES

A	B	C	D	E	F	G
6.08	6.51	8.15	23.09	18.00	15.17	23.00

DISTRIBUTION OF WIND DIRECTION VS STABILITY

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	CALM
A	6	3	1	8	25	12	5	2	8	42	83	25	8	14	13	9	0
B	3	4	8	9	33	24	10	16	31	40	56	24	11	2	5	7	0
C	5	7	10	6	41	19	27	33	41	53	45	29	11	11	9	7	0
D	32	31	39	63	135	68	55	72	91	97	108	74	49	38	26	25	0
E	26	38	37	52	37	22	34	20	35	74	142	114	59	34	29	29	0
F	82	48	36	20	17	10	13	14	16	40	70	56	69	43	42	83	0
G	240	225	130	25	11	8	14	14	16	9	14	23	30	37	80	123	0
TOTAL	394	356	261	183	299	163	158	171	238	355	518	345	237	179	204	283	0

APPENDIX C

DOSE CALCULATIONS

GASEOUS EFFLUENT* DOSE CALCULATION

Doses to the maximum individual and the surrounding population resulting from the release of radioactive material in gaseous effluents from the Palo Verde Nuclear Generating Station were calculated using the GASPAR computer program. Gaseous effluents were released from Units 1 and 2 during the first and second quarters. The radionuclides considered in the dose calculations were Tritium, Iodine-131, Iodine-132, Iodine-133, Iodine-135, all noble gases, and particulates having a half-life greater than eight days and for which dose factors are contained in NUREG-0172. Strontium-89 and Strontium-90 were considered for the first quarter only since the second quarter results were not available. Locations selected for individual dose calculations included for each sector, the site boundary, and within five miles, if present, the nearest residence, the nearest garden and the nearest milk animal. GASPAR implements the radiological dose models of Regulatory Guide 1.109 to determine the radiation exposure to man from four principal atmospheric exposure pathways: plume, ground deposition, inhalation, and ingestion. The ingestion pathways considered were cow milk, goat milk, meat, and vegetables. Doses to the maximum individual and the population were calculated as a function of age group and pathway for significant body organs. Assumptions and data sources used for input to the GASPAR code are described on page C6.

Table C1 presents the doses on a quarterly and semi-annual basis for the highest exposed location on the site boundary and the maximum individual in the general public. The site boundary and residence** locations for which data are presented represent the highest semi-annual doses. Table C1 also presents the total dose as defined in Technical Specification 3.11.4.

Table C2 presents the population doses for the first two quarters, and the first semi-annual period of 1987. Table C3 summarizes the individual doses and compares the result to PVNGS Technical Specification limits.

Based on results obtained by placing TLDs on the site boundary in each sector, the net annual direct radiation dose from both units was determined to be zero.

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*There were no liquid effluents associated with the operation of this facility

**Based on December 1986 land-use census.

TABLE C1

DOSES TO SPECIAL LOCATIONS FOR JANUARY-JUNE 1987

SITE BOUNDARY 1.40 MILES SSW FROM UNIT 1 AND 1.14 MILES SSW FROM UNIT 2

	BETA	GAMMA
AIR DOSES(MRAD)		
1ST QUARTER	8.13E-01	2.85E-01
2ND QUARTER	1.64E-01	6.65E-02
1ST SEMI-ANNUAL	9.77E-01	3.51E-01

	T. BODY	SKIN
MAXIMUM INDIVIDUAL(MREM)		
1ST QUARTER	1.68E-01	4.82E-01
2ND QUARTER	4.01E-02	1.07E-01
1ST SEMI-ANNUAL	2.08E-01	5.89E-01

MAXIMUM INDIVIDUAL IN GENERAL PUBLIC LOCATED AT A RESIDENCE 1.40 MILES N FROM UNIT 1 AND 1.50 MILES NNE FROM UNIT 2

(MREM)	T.BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
1ST QUARTER								
ADULT	6.93E-02	6.94E-02	1.97E-02	6.93E-02	6.93E-02	8.10E-02	6.93E-02	1.06E-01
TEEN	7.59E-02	7.60E-02	1.97E-02	7.59E-02	7.59E-02	8.71E-02	7.59E-02	1.13E-01
CHILD	9.97E-02	9.98E-02	1.97E-02	9.97E-02	9.98E-02	1.15E-01	9.97E-02	1.36E-01
INFANT	2.54E-02	2.54E-02	1.96E-02	2.54E-02	2.54E-02	3.00E-02	2.54E-02	6.21E-02
2ND QUARTER								
ADULT	4.10E-02	4.10E-02	1.45E-02	4.10E-02	4.10E-02	7.08E-02	4.09E-02	6.46E-02
TEEN	4.44E-02	4.45E-02	1.45E-02	4.45E-02	4.45E-02	7.17E-02	4.44E-02	6.81E-02
CHILD	5.71E-02	5.71E-02	1.46E-02	5.72E-02	5.72E-02	9.61E-02	5.71E-02	8.08E-02
INFANT	1.75E-02	1.75E-02	1.45E-02	1.75E-02	1.75E-02	2.45E-02	1.75E-02	4.12E-02
1ST SEMI-ANNUAL								
ADULT	1.10E-01	1.10E-01	3.42E-02	1.10E-01	1.10E-01	1.52E-01	1.10E-01	1.71E-01
TEEN	1.20E-01	1.20E-01	3.42E-02	1.20E-01	1.20E-01	1.59E-01	1.20E-01	1.81E-01
CHILD	1.57E-01	1.57E-01	3.43E-02	1.57E-01	1.57E-01	2.11E-01	1.57E-01	2.17E-01
INFANT	4.29E-02	4.29E-02	3.41E-02	4.29E-02	4.29E-02	5.45E-02	4.29E-02	1.03E-01

TABLE C2

INTEGRATED POPULATION DOSES FOR JANUARY - JUNE 1987

MANREM

JANUARY 1 - MARCH 31 1987

PATHWAY	T.BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
PLUME	4.11E-01	4.11E-01	4.11E-01	4.11E-01	4.11E-01	4.11E-01	4.11E-01	1.46E+00
GROUND	6.17E-05	6.17E-05	6.17E-05	6.17E-05	6.17E-05	6.17E-05	6.17E-05	7.24E-05
INHAL	3.61E-01	3.61E-01	2.48E-04	3.61E-01	3.62E-01	4.63E-01	3.62E-01	3.61E-01
VEGET	1.48E+00	1.48E+00	5.09E-04	1.48E+00	1.48E+00	1.68E+00	1.48E+00	1.48E+00
COW MILK	1.48E-01	1.48E-01	5.64E-05	1.48E-01	1.48E-01	1.69E-01	1.48E-01	1.48E-01
MEAT	5.43E-02	5.43E-02	1.71E-07	5.43E-02	5.43E-02	5.43E-02	5.43E-02	5.43E-02
TOTAL	2.46E+00	2.46E+00	4.12E-01	2.46E+00	2.46E+00	2.77E+00	2.46E+00	3.51E+00
AVERAGE PERSON (REM)	1.37E-06	1.37E-06	2.29E-07	1.37E-06	1.37E-06	1.54E-06	1.37E-06	1.95E-06

APRIL 1 - JUNE 30 1987

PATHWAY	T.BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
PLUME	3.67E-01	3.67E-01	3.67E-01	3.67E-01	3.67E-01	3.67E-01	3.67E-01	1.26E+00
GROUND	5.53E-04	5.53E-04	5.53E-04	5.53E-04	5.53E-04	5.53E-04	5.53E-04	6.51E-04
INHAL	3.18E-01	3.18E-01	6.49E-04	3.18E-01	3.19E-01	5.83E-01	3.18E-01	3.18E-01
VEGET	7.23E-01	7.23E-01	1.04E-03	7.24E-01	7.24E-01	1.12E+00	7.22E-01	7.22E-01
COW MILK	1.34E-01	1.34E-01	1.87E-04	1.34E-01	1.34E-01	2.04E-01	1.34E-01	1.34E-01
MEAT	2.84E-02	2.84E-02	3.76E-07	2.84E-02	2.84E-02	2.85E-02	2.84E-02	2.84E-02
TOTAL	1.57E+00	1.57E+00	3.69E-01	1.57E+00	1.57E+00	2.31E+00	1.57E+00	2.46E+00
AVERAGE PERSON (REM)	8.74E-07	8.74E-07	2.05E-07	8.74E-07	8.74E-07	1.29E-06	8.74E-07	1.37E-06

JANUARY 1 - JUNE 30 1987

PATHWAY	T.BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
PLUME	7.78E-01	7.78E-01	7.78E-01	7.78E-01	7.78E-01	7.78E-01	7.78E-01	2.73E+00
GROUND	6.15E-04	6.15E-04	6.15E-04	6.15E-04	6.15E-04	6.15E-04	6.15E-04	7.24E-04
INHAL	6.79E-01	6.79E-01	8.98E-04	6.80E-01	6.81E-01	1.05E+00	6.80E-01	6.79E-01
VEGET	2.21E+00	2.21E+00	1.55E-03	2.21E+00	2.21E+00	2.80E+00	2.21E+00	2.21E+00
COW MILK	2.81E-01	2.81E-01	2.43E-04	2.82E-01	2.82E-01	3.73E-01	2.81E-01	2.81E-01
MEAT	8.27E-02	8.27E-02	5.47E-07	8.27E-02	8.27E-02	8.28E-02	8.27E-02	8.27E-02
TOTAL	4.03E+00	4.03E+00	7.81E-01	4.03E+00	4.03E+00	5.08E+00	4.03E+00	5.97E+00
AVERAGE PERSON (REM)	2.24E-06	2.24E-06	4.35E-07	2.24E-06	2.24E-06	2.83E-06	2.24E-06	3.32E-06

Table C3
SUMMARY OF INDIVIDUAL DOSES FOR JANUARY - JUNE 1987*

	Unit	Quarter #1	Quarter #2	Year to Date
Gamma Air Dose	mrad	2.85E-01	6.65E-02	3.51E-01
T.S. 3.11.2.2 Limit	mrad	5.00E+00	5.00E+00	1.00E+01
% T.S. Limit	%	5.70E+00	1.33E+00	3.51E+00
Beta Air Dose	mrad	8.13E-01	1.64E-01	9.77E-01
T.S. 3.11.2.2 Limit	mrad	1.00E+01	1.00E+01	2.00E+01
% T.S. Limit	%	8.13E+00	1.64E+00	4.89E+00
Maximum Organ Dose (neglecting skin)	mrem	1.15E-01**	9.61E-02**	2.11E-01
T.S. 3.11.2.3 Limit	mrem	7.50E+00	7.50E+00	1.50E+01
% T.S. Limit	%	1.53E+00	1.28E+00	1.41E+00

*From Table C1.

**These control location doses are imparted via four principal atmospheric pathways: plume, ground exposure, inhalation and vegetable ingestion. The hypothetical site boundary location doses presented in Tables A4 and A8 of Appendix A, are based on the ODCM, and also include milk and meat ingestion contributions. The highest organ dose is to the child's thyroid resulting from exposure in the North Sector at 1.4 miles, and the North Northeast Sector at 1.5 miles, from Units 1 and 2, respectively. Technical Specification 3.11.4 has higher limits than Technical Specification 3.11.2.3 and therefore the percent of limits are more conservative based on Technical Specification 3.11.2.3 than on Technical Specification 3.11.4.

DOSE CALCULATION MODELS

The GASPAR computer code was used to evaluate the radiological consequences of the routine release of gaseous effluents. GASPAR implements the dose calculational methodologies of Regulatory Guide 1.109, Revision 1.

Source terms for each quarter are combined with station-specific demographic data and each quarter's atmospheric diffusion estimates for gaseous dose calculations.

Atmospheric diffusion estimates are generated by the XOQDOQ computer code using onsite meteorological data as input. Doses for the semi-annual period are the summation of the quarterly doses. Additional input to GASPAR includes the following site specific data:

- 0 to 5 mile land use census conducted in December 1986. The following changes in land use were identified between December 1985 and December 1986; addition of a goat location within the East-Northeast sector and 4.75 miles from Unit 2 and a residence located within the East sector and 2.25 miles from Unit 2.
- 0 to 5 mile population distribution based on the land use census conducted June-August, 1984.
- 5 to 50 mile population distribution from PVNGS ER-OL Figure 2.1-6.
- The population distribution of metropolitan Phoenix greater than 50 miles from PVNGS, based on the 1980 census results, were conservatively included in the 40 to 50 mile sectors.
- Absolute humidity of 6.0 g/m^3 from PVNGS ER-OL Table 2.3-34.
- The fraction of the year that vegetables are grown (0.667) from PVNGS ER-OL Section 2.1.3.4.
- The fraction of daily feed derived from pasture while on pasture and length of grazing season for milk animals beyond 5 miles (0.35 and 0.75) from PVNGS ER-OL Section 2.1.3.4.
- The only milk animal (goat) located within 5 miles from PVNGS was fed entirely on stored feed during 1986, based on the land use census conducted in December 1986.
- The fraction of daily feed derived from pasture while on pasture and length of grazing season for meat animals (0.05 and 0.25) from PVNGS ER-OL Section 2.1.3.4.

Other values used for input to GASPAR are default values from Regulatory Guide 1.109 Revision 1.

APPENDIX D

PROCESS CONTROL PROGRAM (PCP)

REVISION 1

CONTROLLED BY USER

SOLID RADWASTE

PROCESS CONTROL PROGRAM

PALO VERDE NUCLEAR GENERATING STATION

ASSIGNED COPY

PVNGS # 2

Department Head:	Thomas Hillmer	Date:	1/6/87
PRB Chairman:	[signature]	Date:	3/23/87
Plant Manager:	[signature] for	Date:	3/30/87
Effective Date:	~~4/14/87~~ 4-1-87 4/3/87		

REV 1

CONTROLLED BY USER

TABLE OF CONTENTS

APPENDICES

1.0 PURPOSE

The purpose of the Process Control Program (PCP) for the Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3 is to establish a set of process parameters which will provide reasonable assurance of complete solidification of various liquid radioactive "wet wastes" including resin slurries, evaporator bottoms, filter sludges, and chemical drains in accordance with the requirements of applicable portions of the PVNGS Quality Assurance Program, PVNGS Technical Specifications, PVNGS Final Safety Analysis Report, Department of Transportation (DOT) regulations, Nuclear Regulatory Commission (NRC) regulations, and licensed burial facilities acceptance criteria for solidification, packaging and shipment to an approved offsite burial site. There are no Arizona State Regulations applicable to PVNGS.

Toward this purpose, the PCP ensures that the solidified substance is a monolith having no freestanding liquid and is within the limits as set forth in the above mentioned regulations and acceptance criteria. This PCP will also ensure that solidification will be performed to maintain any potential radiation exposure to plant personnel to "as low as is reasonably achievable" (ALARA) levels, in accordance with the "ALARA Program" procedure.

2.0 DEFINITIONS AND ABBREVIATIONS

a. ALARA -As low as is reasonably achievable.

b. Batch -A quantity of water type(s) prepared in the waste feed tank for solidification.

c. Bench Test -Laboratory testing of the solidification process on a reduced scale with representative mixing ratios and similar operational techniques.

d. Wet Waste Types -Liquid radioactive wastes, sludges, and ion exchanger resins.

e. Monolith -A freestanding, solid object.

f. Procedure -A document that specifies or prescribes how an activity is to be performed. Procedures shall be reviewed and approved by the appropriate Department Head prior to implementation.

g. Department-Directive A category of document below the level of policies, procedures, and programs that specifies policies, activities, or systems unique to a department. Directives will be used to implement licensing, or

Rev. 01

Rev. 01

regulatory requirements, or task which directly effect quality related activities or equipment which are not covered by plant procedures. Review and approval shall be in accordance with Technical Specifications.

3.0 PROGRAM DESCRIPTION

3.1 Waste Types

a. Appendix A, "Schematic Flow Diagram," and Appendix B, "Radwaste Cement Solidification System Diagram," illustrate the different waste types to be processed by the solidification system. They are listed as follows:

1) Chemical regenerative and decontamination waste (evaporator concentrates) from a forced recirculation evaporator.

2) Boric acid waste from the Boric Acid Concentrator.

3) Bead resin waste as a slurry from the Spent Resin Tanks.

4) Chemical Drain Tank waste.

5) Spent filter cartridges.

b. Typical and atypical wet waste types shall be identified by the "Solidification Process Control" procedure, which provides a record of waste formulations.

c. When the radiation level of a waste batch is determined to be excessive by ALARA standards, a waste substitute shall be prepared in accordance with the "Solidification Process Control".

3.2 Process Parameters

a. An acceptable waste product for transfer and disposal in accordance with 10 CFR 20.311 for class A, B, and C waste shall be provided by compliance with the "Solidification Process Control" procedure and "Classification of Radioactive Waste" Radwaste directive.

Rev. 01

b. The process parameters for various class-A wastes shall be based on the Hittman Radwaste Solidification System (Cement) Topical Report, HN-R1109, Revision 4 or U.S. Gypsum Envirostone Topical Report 5/84, or Chem-Nuclear Systems, CNSI-WF-C-02-P, (or a vendor process control program) and the "Solidification Process Control" procedure. These documents establish boundary conditions to provide reasonable assurance that solidification will be complete.

Rev 01

c. For class-A waste types containing concentrations of chemicals that do not fall within the bound of chemical concentrations for which preoperational solidification tests have been performed by Hittman Nuclear and Development Corporation, or U.S. Gypsum, or Chem-Nuclear Systems acceptable base data for test solidificaitons shall be developed in accordance with the "Solidification Process Control" procedure.

d. As plant conditions dictate, including ALARA considerations as well as inplant system inoperability due to maintenance, repairs, or modifications a portable solidification system will be used to process class A waste in accordance with the vendor's operating procedures. Class B and C waste will be processed in accordance with the vendor's operating procedures and a NRC approved 10 CFR 61 Topical Report.

e. For class-B and class-C waste types, a 10 CFR 61 qualified solidification process will be used on the installed solid radwaste system in accordance with the "Solidification Process Control" procedure or a portable solidification system will be used in accordance with the vendor's operating procedures and a NRC approved 10 CFR 61 Topical Report.

f. Process mixing ratios for class-A, B, and C waste types shall be determined for each waste batch in accordance with the "Solidification Process Control" procedure.

3.3 Waste Classification

Rev.01

a. The "Waste Stream Scaling Factors" department directive shall provide PWR scaling factors for identifying specific radionuclides as required by 10 CFR 61.55, "Waste Classification".

b. Scaling factors shall be verified or updated as required in 10CFR Part 61 by waste stream analysis to ensure acceptable standards are maintained for waste classification.

Rev. 01

c. During incident conditions where the use of the existing scaling factors is questionable, the waste shall be classified by correlation factors or actual sample analysis in accordance with the "Classification of Radioactive Waste" directive.

3.4 Waste Preconditioning

a. The "Operation of Solidification System" procedure shall designate the required mixing/recirculation times and system operations to ensure a representative sample is obtained after chemical addition and prior to process initiation.

b. Adjustment of the waste solution pH shall be in accordance with the "Operation of Solidification System" procedure.

c. The "Operation of Solidification System" procedure shall designate when heat tracing is required to ensure chemical suspension.

3.5 Verification of Solidification

a. Solid Radwaste System

Rev. 01

1) The solidification bench tests, in accordance with the Hittman Topical Report, Radwaste Solidification System (Cement), HN-R1109, Revision 4, or U.S. Gypsum Envirostone Topical Report 5/84, or Chem-Nuclear Systems (CNSI-WF-C-02-P) provide the solidification bench testing base data for the "Solidification Process Control" procedure for use with class-A type waste.

2) The solidification bench tests in accordance with a vendor's NRC approved 10 CFR 61 Topical Report will provide the bench testing base data for the "Solidification Process Control" procedure for use with class-B and C type waste.

3) During solidification system operations, additions to the waste feed tank, waste feed pump operation, and process mixing ratio adjustments shall be in accordance with the "Operation of Solidification System" procedure.

   i. No additions will be made to the waste feed tank while the tank is being recirculated for sampling or processing.

   ii. The waste feed pump will not be stopped during sampling.

4) A periodic solidification bench test shall be performed as specified by and in accordance with the "Solidification Process Control" procedure for verification of an acceptable solidification process.

5) If any solidification bench test is found to be not acceptable, subsequent operations and testing shall be in accordance with the "Solidification Process Control" procedure.

6) The solidification bench test acceptance criteria shall be in accordance with the "Solidification Process Control" procedure.

7) Solidification product quality is controlled by the performance of the "Operation of Solidification System" procedure.

b. Portable Solidification System

The portable solidification vendor will verify proper solidification of the waste product in accordance with the vendor's operating procedures and a NRC approved 10 CFR 61 Topical Report.

c. Handling of containers of unacceptable solidified waste shall be in accordance with the "Operation of Solidification System" procedure.

3.6 Stability Requirements

The "Solidification Process Control" procedure shall identify the stability specifications as set forth in 10 CFR 61.56, "Waste Characteristics," and Branch Technical Position ETSB 11-3, Revision 2, July 1981, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants."

3.7 Data Sheets

a. For each solidification bench test actually used for waste processing, a test data record shall be maintained in accordance with the "Solidification Process Control" procedure.

b. For each batch solidification process, a feed rate determination shall be completed in accordance with the "Solidification Process Control" procedure.

c. For each batch solidification process, records shall be maintained of the unique batch information in accordance with the "Solidification Process Control" procedure.

Rev. 01

d. For each batch solidification process, a waste classification record shall be completed in accordance with Radwaste Directive No. 13, "Classification of Radioactive Waste".

3.8 Record Retention

For each batch solidification process, all records generated shall be maintained in accordance with the "Document/Record Turnover Control" procedure.

3.9 Radiological Precautions

The radiological precautions necessary for implementing the Process Control Program shall be followed and are covered in the "Radiation Protection Program" procedure.

Rev. 01

REFERENCES

a. NUREG-0472, "Radiological Effluent Technical Specifications for PWRS".

b. 10 CFR 20, "Standards for Protection Against Radiation."

c. 10 CFR 71, "Packaging and Transportation of Radioactive material."

d. Palo Verde Nuclear Generating Station Technical Specifications.

e. 49 CFR Subchapter C - "Hazardous Materials Regulations."

f. Palo Verde Nuclear Generating Station Final Safety Analysis Report, Sections 11.4, 12.1, and 12.3.

Rev. 01

g. Palo Verde Nuclear Generating Station program, "ALARA Program," 75PR-9ZZ03.

h. Palo Verde Nuclear Generating Station Final Safety Analysis Report, Section 17.2, "Quality Assurance During the Operating Phase."

i. Palo Verde Nuclear Generating Station manual, "Operations Quality Assurance Criteria Manual".

Rev. 01

j. Palo Verde Nuclear Generating Station procedure, "Solidification Process Control," 75RP-9SR01.

k. Palo Verde Nuclear Generating Station procedure(s), "Operation of Solidification System," 75OP-(1,2,3)SR02.

l. Palo Verde Nuclear Generating Station Radwaste Directive, "Classification of Radioactive Waste," RWD-019.

m. ANPP Administrative Procedure, 9N219.50.00.

n. USNRC Branch Technical Position-ETSB 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants."

o. 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste."

Rev. 01

p. Palo Verde Nuclear Generating Station program, "Radiation Protection Program." 75PR-0ZZ01.

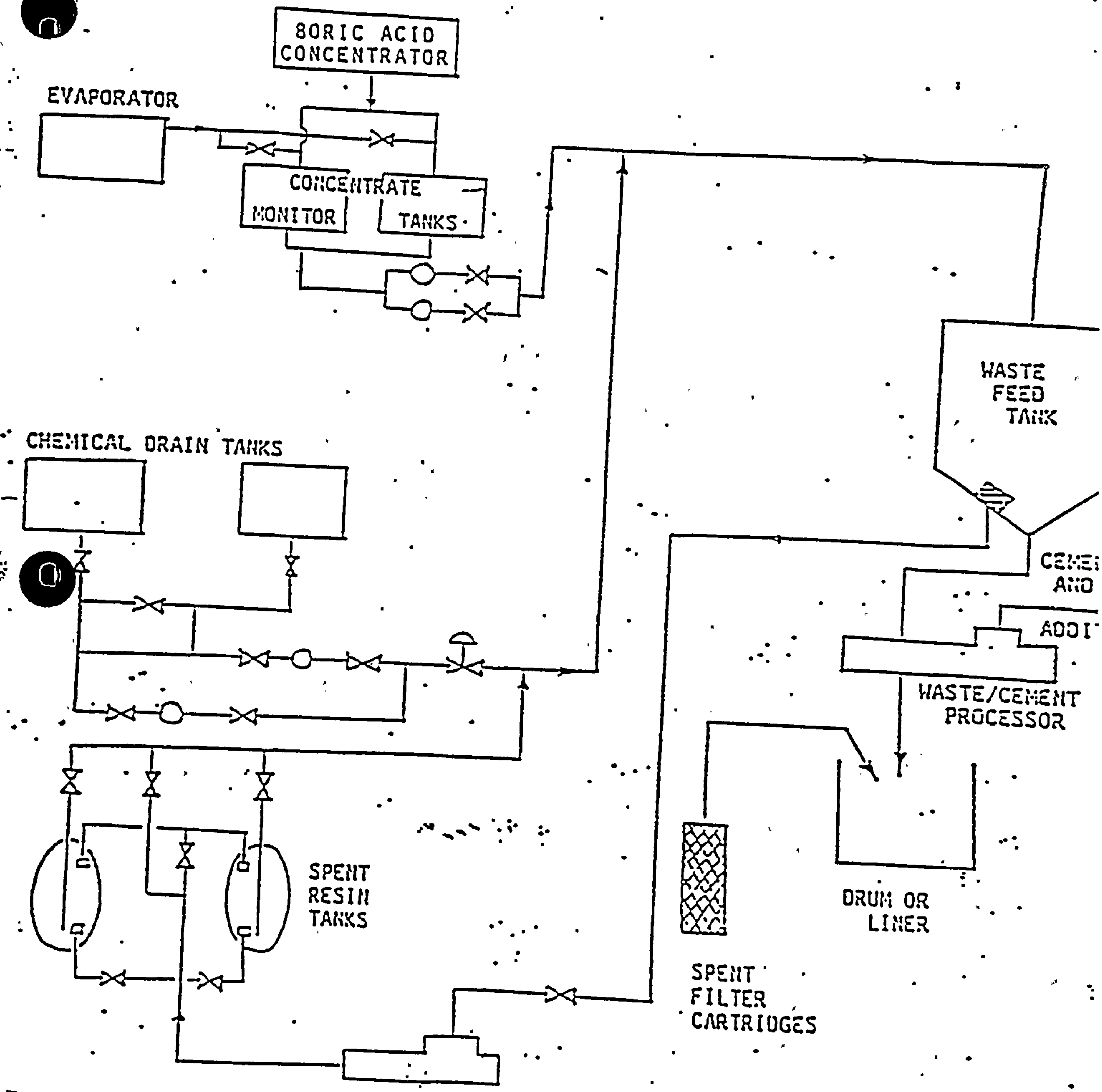
q. U.S. Gypsum Envirostone Topical Report 5/84.

r. Chem-Nuclear Systems (CNSI-WF-C-02-P), "Develooment and Testing of Waste Solidification Formulas to meet Title 10 CFR Part 61 Waste Form Criteria".

s. Palo Verde Nuclear Generating Station Radwaste Directive, "Waste Stream Scaling Factors," RWD-012.

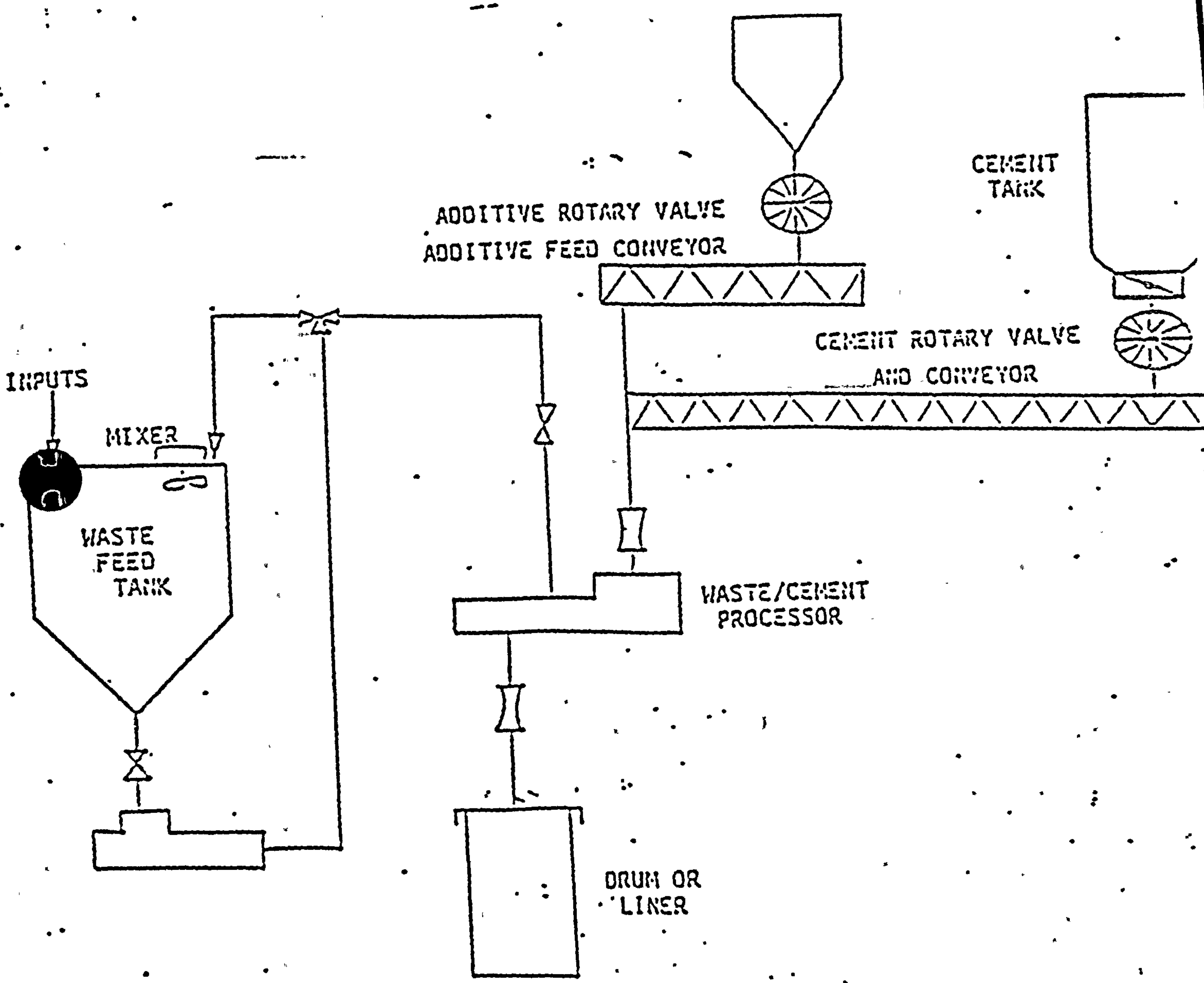
t. Palo Verde Nuclear Generating Station Radwaste Directive", "Aquaset, Aquaset II, Petroset I, and Petroset II Absorption Process," RWD-010.

APPENDIX A



SCHEMATIC FLOW DIAGRAM

APPENDIX 8



RADWASTE CEMENT SOLIDIFICATION SYSTEM DIAGRAM

KRB

ARIZONA NUCLEAR POWER PROJECT
Post Office Box 21666 Phoenix, Arizona 85036

received 9/19/84

JND

September 14, 1984
ANPP-30516

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: PVNGS Units 1, 2, and 3
Proof and Review Technical Specifications
Docket Nos. STN 50-528/529/530
File: 005-419.05

Dear Mr. Knighton:

Your letter dated August 14, 1984 transmitted to APS our copy of the PVNGS proof/review Technical Specifications. Your letter requested us to review and respond to our proof and review technical specifications. Due to the detail of our review we are submitting our response to you one day late. This was discussed with M. Licitria, D. Brinkman (NRC) and S. R. Frost (APS). The one day did not present any problems for the reviewers.

In performing our PVNGS Technical Specification Review we developed a committee to review and comment on the NUREG 0212 Rev. 3 approximately two years ago. Our committee consisted of offsite engineering, Licensing, onsite Operations, H.P./Chemistry, Maintenance, Engineering, Startup, QA, STA/ISEG, I and C, Training, Bechtel Engineering and Combustion Engineering. This committee worked closely with the NRC reviewer to develop a set of technical specifications that represented PVNGS.

This committee functioned, as follows, to mold the CE Standard Tech Specs so they would not only represent the design of PVNGS but also represent how the plant will be operated:

1) Utilize our own Plant Specific experience to review systems, their functions, parameters and system names.

2) Discussed Tech Spec problems with operating units throughout the industry.

3) Held review meetings with various operating units.

4) Had operating experienced units review/comment on our proof/review tech specs.

5) We have used our Tech Specs during our startup program to see if we can live with the various specs and associated equipment in order to eliminate future problems (i.e., pump performance etc.).

6) Monitored Federal Register to see if any Tech Spec changes other plants obtained would applying to PVNGS.

7) Reviewed various operating experiences (i.e., LERs, some inspection reports, etc.) to see if they could affect the Tech Specs.

541 pp.
~~8411020130~~

8) Compared the Tech Specs to the PVNGS FSAR for consistency.

9) Compared the Tech Specs to the PVNGS SER for consistency.

10) Compared the Tech Specs to the CESSAR FSAR for consistency.

11) Compared the Tech Specs to the CE-SER for consistency.

12) We have used our vendor's experience and support from the beginning to develop our Tech Spec.

13) We have had our DCPs reviewed to see if Tech Spec changes are needed.

14) We have trained our operators in our "marked up" Tech Specs over the past 2 years.

15) We have utilized our Tech Specs on the PVNGS Plant Specific Simulator.

16) We have monitored/solicited questions and interpretation problems from Training and Operations and revised our Tech Specs to make the Tech Specs clear for everyone.

17) We have written our procedures from our marked up Tech Specs and as problems arise we may have changed the spec.

18) Continuous discussions over the past two years with our resident inspectors and resolving their problems either through discussion or revision to the Tech Spec.

We believe that we have conducted a detailed review of the PVNGS Tech Specs and have a good operational document if issued in a final form as we have amended Attachment A. All of our changes marked in the proof/review copy have justifications in Attachment B. Many of the changes that are identified in this marked up proof/review copy have been submitted along with their justifications over the past years.

We feel very strongly that we need all of the attached changes for the following reasons:

1) This is how we will operate the unit.

2) Some of the changes are a "human factors" consideration that will hopefully eliminate errors that other operating plants have experienced.

3) To avoid massive amount of Tech Spec changes after we go operational (as experienced by other utilities).

The new NRC Tech Spec program requires that the licensee certify their Tech Specs prior to final acceptance. It is our position that in order to certify the Tech Specs that they not only have to reflect the design of PVNGS but also its operation. Therefore, we will need to implement all the changes identified in Attachment A to this letter.

If you have any questions please contact S. R. Frost (602) 943-7200, extension 6183.

Very truly yours,

E E Van Brunt /RSK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/SRF/wpc

cc: E. Licitra (w/a)
J. B. Martin (w/a)
R. Zimmerman (w/a)
G. Fiorelli (w/a)

PG 3/4 9-12 DELETE: me

See page.

JUSTIFICATION:

Typo.

PG 3/4 9-13 CHANGE: ASB AEB

LCO. See page.

JUSTIFICATION:

The correct number (22 feet 8 inches) of water shall be maintained over the top of the storage racks is the correct LCO. The 22 feet 8 inches is needed to ensure the minimum water depth to remove a nominal 99% of the assumed gap activity released from a ruptured irrigated fuel assembly lying on its side on top of the storage racks.

PG 3/4 10-6 CHANGE: RSB. me

LCO Item C. See page.

JUSTIFICATION:

We want to add "Key-Locked" to LCO Item C. This depicts the actual way we will operate and maintain the valves of the Safety Injection tanks to be open.

PG 3/4 10-8 CHANGE: RSB

LCO Item b. See page.

JUSTIFICATION:

The addition of ..."or not to go below 254 psig" is needed to alert the operator of this operating limit for this special test exception.

PG 3/4 11-2 CHANGE: RAB METB

Table 4.11-1. See page.

JUSTIFICATION:

Number left off Table.

PG 3/4 11-3 CHANGE: RAB METB

Footnote b. See page.

PROOF AND REVIEW

METB

JUSTIFICATION:

PVNGS does not have a method to thoroughly mix liquid wastes the way it is presently stated in the Proof/Revision Tech. Spec.

Footnote b was rewritten to comply with the way PVNGS will perform this function.

PG 3/4 11-5 DELETE:

RAS

Tech. Spec. Request the deletion of this specification, its basis B 3/4 11.1.2, and removal of reference to liquid radioactive effluent releases with paragraph 2, 3, 6, and 8 Technical Specification 6.9.1.8.

JUSTIFICATION:

There are no liquid release points from which activity is discharged for use by the general public. In addition, sampling required by Technical specification Table 4.11-1 will monitor liquid discharges to the evaporation pond. When this Tech spec (3/4 11.1.2) was initially requested to be changed the reviewer developed Tech. Spec. 4.11-1 for our evaporation pond so 3/4 11.1.2 could be deleted since we are a zero release plant.

PG 3/4 11-6 CHANGE:

METB

LCO. See page.

JUSTIFICATION:

BPC recommended we change the 10 Ci to 500 Ci in this Tech Spec. This change is supported by the PVNGS FSAR, Section 2.4-13 (page 2.4-118). For this accident BPC assumed a 100 Ci event multiplied this number by 5 and you still are below the required MPC limit; therefore, the change to 500 Ci is acceptable and supported by the FSAR and will not exceed the MPC allowable limit.

11-7
11-8
PG 3/4 11-10 CHANGE:

RAS
METB

Footnote b. See page.

JUSTIFICATION:

PWR's do not vent off gas directly to atmosphere as do BWR's. Any increase in gaseous activity due to power level change would be the result of increases in the fuel leakage rate to the primary coolant. Primary system valve leakage to containment atmosphere or to the auxiliary building atmosphere would be the sources of

PG 3/4 11-4

METB

METB

discharge to the environment. Both of these discharge pathways pass through the plant vent. An increase in the noble gas monitor count rate of greater than a factor of 3 would indicate that the isotopic mixture of the effluent had change sufficiently to warrant a re-analysis of alarm setpoints and projected offsite dose. Grab samples of noble gas and Tritium are analyzed weekly for the generation of Gaseous Effluent Release Permits. It is felt that the weekly sampling frequency in addition to the Containment Atmosphere Monitor readings will monitor plant vent discharges so they will remain below Technical Specification and 10CFR20 limits. The Fuel Building Exhaust discharge should not be affected by power level changes.

PG 3/4 11-15 CHANGE: METB

Surveillance 4.11.2.6. See page.

JUSTIFICATION:

Add ... "During Addition" to the Surveillance Requirement. This is needed so it identifies how we will perform this surveillance more accurately thus, avoiding possible errors. Since this spec is for quantity of radioactivity in the gas storage tank, it should only be monitored during additions to that tank.

PG 3/4 11-17 CHANGE: METB

LCO. See page.

JUSTIFICATION:

PVNGS does not have filter sludges; therefore, this reference in the LCO is not applicable to the PVNGS design.

PG 3/4 11-18 CHANGE: RAB

Surveillance 4.11.4.1. See page.

JUSTIFICATION:

Delete all references to liquid effluent for the justification stated above for 3.11.1.2.

PG 3/4 5-6 CHANGE: ASB ~~ETS~~

5.6.1.1.b.

~~B3/4 1-1~~ ~~ETS~~

PG 5-7 5-8 CHANGE:

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

RSB
CPB

CH6 CHANGE:

See page.

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

→ RAB See next page

BASIS CHANGES

See pages.

JUSTIFICATION

The basis has been revised in pacts to more accurately reflect our plant.

See next page

pages 6-13 to 6-25 as appropriate

Bases

Pg	Branch(s)
B 3/4 1-1	CPB
B 3/4 1-2	CPB
1-3	CPB
2-1	CPB
3-2	METB
3-3	CEB
3-4	ASB
4-1	RSB
4-6	RSB MTEB
4-7	RSB MTEB
4-11	RSB MTEB
5-1	CPB
5-3	RSB
6-2	CSB
6-3/4	AEB CSB

Pg	Branch(s)
B3/4 7-1	RSB ASB
7-2	RSB ASB
7-3	ASB
7-4	MEB
7-6/7	CEB
9-3	AEB ASB
10-1	CPB
10-2	RSB
11-1/2	RSB METB
11-5	METB

Justification from applicant for the following proposed changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications:

Table 3.3 - 13, Page 3/4 3-63	METB
Bases Section 3/4 3.3.1, Page 3 - 2	METB
3/4.11.2.1, Page 3/4 11 - 7	RAB/METB
Table 4.11-2, Page 3/4 11-10	RAB/METB
Table 4.4-4, Page 3/4 4-26	METB
3.11.2.5.b, Page 3/4 11-14	METB

MTEB → Table 3.3-13 Page 3/4 3-63

1. Applicability designators * and ** along with their descriptions must be left in as they refer to the Plant Vent Monitor and the Explosive Gas Monitoring System respectively.

2. Action 37 must be changed to read the following:

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days, provided the actions of "A" or "B" are performed.

A. Initiate the preplanned alternate method of sampling the appropriate parameter(s).

B. Take grab samples at least once per 12 hours and analyze for gross activity within 24 hours.

Current Justification

Justification: This allows for ~~the~~ the implementation of the preplanned alternate sampling program allowed for effluent monitors under the former Action Statement 27 of Table 3.3-6

Action Statement 37 as amended at Proof and Review Approval session is unacceptable as it imposes a ~~[illegible]~~ 14 Reporting Requirement where it was only

necessary to explain inoperability of monitors in the Semi Annual Effluent 1.21 Report. (prior to meeting)

Monitoring for effluent noble gas concentrations may be accomplished by grab sampling at a specified frequency. Monitor installation should not be a requirement when a monitor is inoperable.

There are no process monitors listed in this Table this part of the Action 37 refers to the Containment Atmosphere monitor and should not be listed here as it is in Table 3.3-6 Action 27

MTEB ~~Ttl~~ Bases under Section 3/4 3.3.1 Page B3/4

B 3/4 3-2

Delete (3)! We are committed to requirements of NUREG 737 and Reg Guide 1.97 already no need for this in Tech. Specs. Already in 6.8.1.a under Admin Controls

5 of 7

RAB
MTEB

3/4.11.2.1 Page 3/4.11-7

Change to:

6. For [Iodine-131, Iodine 133,] Tritium and for all radio....

Justification: These are only isotopes of Iodine of consequence to Off-site dose due to gaseous effluent releases as described in NUREG 0177 others would be of consequence perhaps in liquid release or be considered in Solidified Waste Part 61 requirements. This will allow agreement with Tech Spec 3.11.2.3.

3/4 11-7

4.11.2.1.2

Change to:

The Dose Rate due to Iodine-131, Iodine-133, Tritium, and all radionuclides ~~of~~ in part. form with half lives greater than 8 days in gaseous effluents... →

RAB
MTEB

Table 4.11-2 Page 3/4 11-10

Delete item "d" as this is not referenced in the Table and sampling frequency is ridiculous. This parameter is ~~___~~ sampled at the Plant vent monthly ~~which~~ This should satisfy reporting requirements as tritium equilibrium in the refueling water and coolant would not vary significantly over this period of time

3/4 11-10

Justification

Frequency of Fuel Building Tritium sampling item "f" should be monthly also. for reason given above.

Admin Control
6.8.1.j
add: "Preplanned Alternate Monitoring and Sampling Program" to list under PRB control.
This implements alternatives ~~changes~~ to monitor and sampling scedules in Table 3.3-6 and Table 3.3-13

METB Table. 4.4-4 Page 3/4 4-26

Add qualifier to 4.6 ** "One sample is sufficient if plant has ~~come to a~~ gone through SCRAM condition or if transient is complete in 6 hours"

Justification: Interpretation is possible that more than one sample is required if power level varies greater than 15%

Table 4.11-2 Page 3/4 11-10

change to

item 6. ... If noble gas monitor is inoperable, samples must be obtained As soon as possible and must be analysed within a Four hour period. This requirement does not apply to the Fuel Building ~~Vent~~ Exhaust

METB 3.11.2.5.b Page 3/4 11-14

change "within 1 hour" to "within 6 hou[rs]"

Justification: Release rate is limited t[o] 50 cfm and average release of tank will take 6 hours

3/4 11-14

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3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGES TO ONSITE EVAPORATION PONDS CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material discharged from secondary system liquid waste to the onsite evaporation ponds shall be limited to the lower limit of detectability (LLD) defined as 5×10^{-7} μCi/ml for the principal gamma emitters or 1×10^{-6} μCi/ml for I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

When any secondary system liquid waste discharge pathway concentration determined in accordance with the surveillance requirements given below exceeds the specified LLD, divert that discharge pathway to the liquid radwaste system without delay.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes collected in the chemical waste neutralizer tank shall be sampled and analyzed prior to their batchwise discharge to the onsite evaporation pond in accordance with the sampling and analysis program specified in Table 4.11-1.

4.11.1.1.2 With the concentration of radioactive material in the chemical waste neutralizer tank exceeding the specified LLD, sample and analyze other secondary system discharge pathways in accordance with the sampling and analysis program specified in Table 4.11-1.

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TABLE 4.11-1

SECONDARY SYSTEM LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

SECONDARY SYSTEM LIQUID RELEASE PATHWAY	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/mL)
A. Batch discharges[b]				
1. Chemical Waste Neutralizer Tank	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
2. Steam Generator Blowdown Low TDS Sump*	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
3. Condensate Polishing Low TDS Sump*	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
B. Continuous Releases[d]				
1. Turbine Building Sump*	D Grab Sample	D Grab Sample	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
2. Condenser Area Sumps*	D Grab Sample	D Grab Sample	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}

*Sampling and analysis for pathways 2 and 3 under batch discharges and 1 and 2 under continuous releases are required only when concentration for chemical waste neutralizer tank pathway exceeds the LLD.

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TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

b PRIOR TO DISCHARGE EACH BATCH SHALL BE ISOLATED AND SAMPLED IN A REPRESENTATIVE MANNER

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TABLE 4.11-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[d]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-3) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

DELETE

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

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3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site (see Figures ~~5.1-1~~ and 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For I-131 AND I-133 ~~all radioiodines,~~ for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to ~~radioactive materials, other than noble gases~~, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

I-131, I-133, TRITIUM AND ALL RADIONUCLIDES IN PARTICULATE FORM WITH HALF-LIVES GREATER THAN 8 DAYS

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[h] G	1×10^{-4}
B. Containment Purge	P Each Purge[b,c] Grab Sample	P Each Purge[b,c]	Principal Gamma Emitters[h] G	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Vacuum Pump Exhaust 2. Plant Vent 3. Fuel Bldg. Exhaust	M[b,e] E Grab Sample	M[b]	Principal Gamma Emitters[h] G	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[g] F	W[d] D Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[g] F	W[d] D Particulate Sample	Principal Gamma Emitters[h] G (I-131, Others)	1×10^{-11}
	Continuous[g] F	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[g] F	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
D. All Radwaste Types as listed in A., B., and C. above.	Continuous[g] F	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

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TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is the smallest concentration of radioactive material in a sample that will yield a net count above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detectable concentration, as the detection capability for a given instrument procedure and type of sample.

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between the midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:

(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report (ARH-2537 (June 22, 1972).

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TABLE 4.11-2 (Continued)

TABLE NOTATION

b~~Analyses shall also be performed following SHUTDOWN, STARTUP, or a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.~~

cSampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

~~dTritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.~~

D ~~e~~Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). ~~Sampling shall also be performed at least once per 24 hours for at least 7 days following each SHUTDOWN, STARTUP, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing. [When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.]~~ STET

E ~~f~~Tritium grab samples shall be taken at least MONTHLY ~~once per 7 days~~ from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

F ~~g~~The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

G ~~h~~The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

b ANALYSES SHALL ALSO BE PERFORMED FOLLOWING SHUTDOWN, STARTUP, OR A THERMAL POWER CHANGE EXCEEDING 15% OF THE RATED THERMAL POWER WITHIN A 1 HOUR PERIOD UNLESS 1) ANALYSIS SHOWS THAT THE DOSE EQUIVALENT I-131 CONCENTRATION IN THE PRIMARY COOLANT HAS INCREASED MORE THAN A FACTOR OF 3; AND (2) THE NOBLE GAS ACTIVITY MONITOR ON THE PLANT VENT SHOWS THAT EFFLUENT ACTIVITY HAS INCREASED BY MORE THAN A FACTOR OF 3. IF THE ASSOCIATED NOBLE GAS VENT MONITOR IS INOPERABLE, SAMPLES MUST BE OBTAINED AS SOON AS POSSIBLE ANALYSES SHALL BE PERFORMED WITHIN A FOUR HOUR PERIOD. THIS REQUIREMENT DOES NOT APPLY TO THE FUEL BUILDING EXHAUST.

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in gaseous effluents exceeding twice the limits of Specifications 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.11.4a.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Director of Nuclear Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the Control Room, a designated SRO, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1. or as specified in the FSAR.

UNIT STAFF

6.2.2.1 The unit organization shall be as shown in Figure 6.2-2 or as specified in the FSAR and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A radiation protection technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the (2) other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

6.2.2.2 The unit staff working hours shall be as follows:

a. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

---

*The radiation protection technician and Fire Team composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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UNIT STAFF (Continued)

b. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed (THIS EXCLUDES THE STA WORKING HOURS):

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

c. Any deviation from the above guidelines shall be authorized by the Director of Nuclear Operations or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Director of Nuclear Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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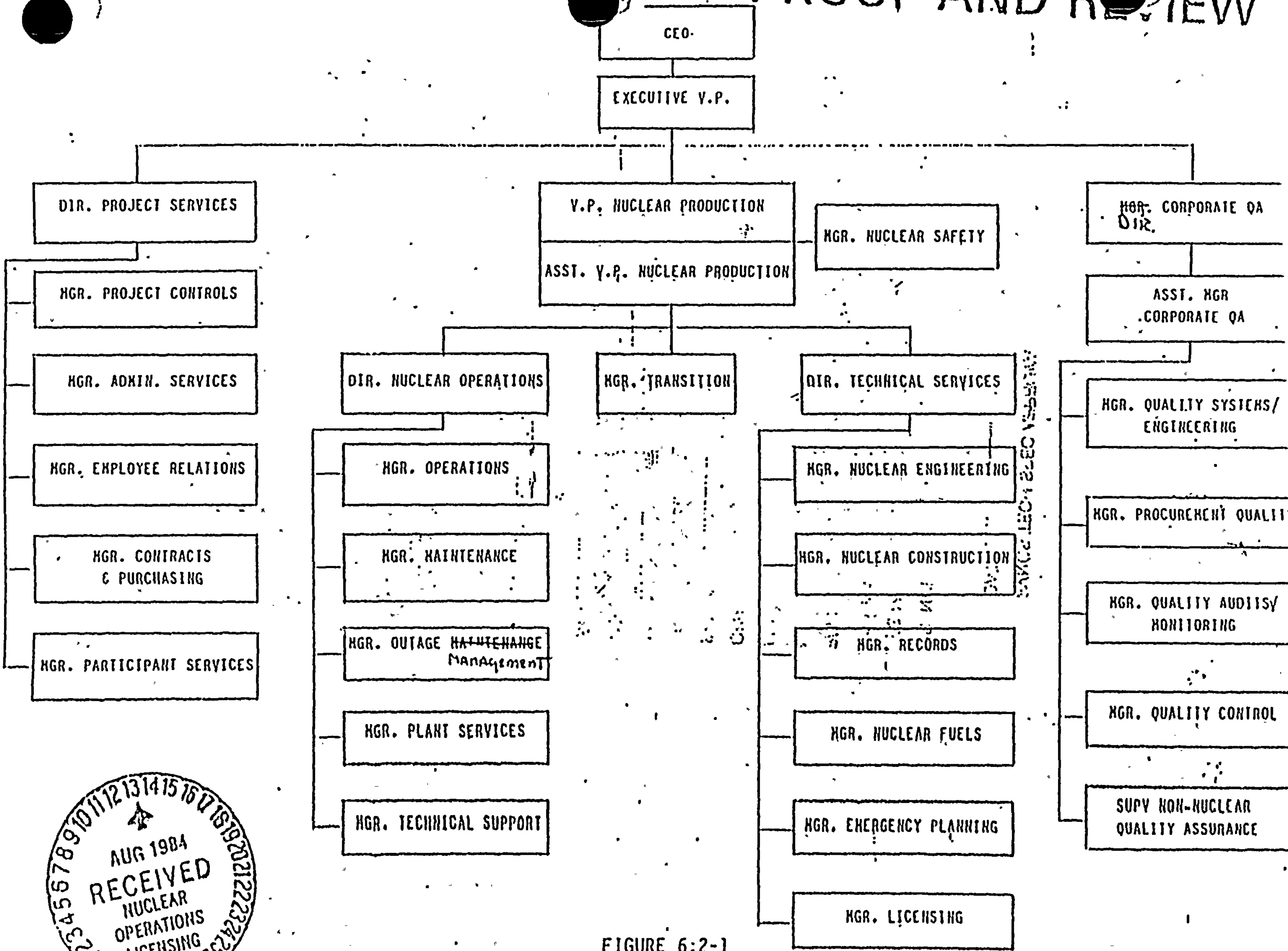

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FIGURE 6:2-1
OFFSITE ORGANIZATION

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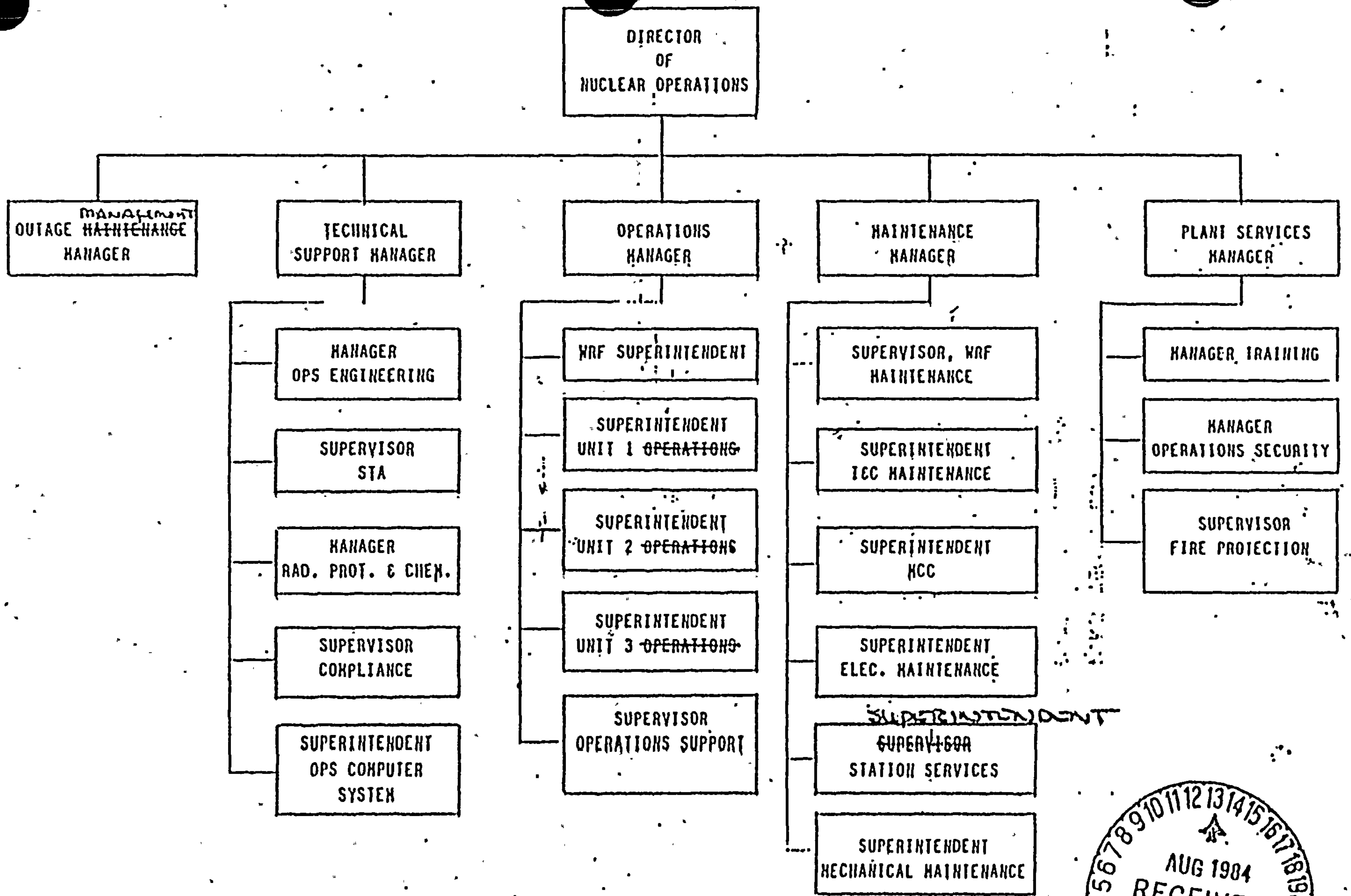

FIGURE 6.2-2

ONSITE UNIT ORGANIZATION

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Nuclear Operator I or II
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the Control Room command function.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 ~~The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.~~

STET / See Attachment 6-F

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, Director of Nuclear Operations, and the Supervisor, Nuclear Safety Group (NSG).

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager of Nuclear Safety. ~~and Supervisor of the NSG.~~

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1-1978, as endorsed by Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

*Not responsible for sign-off function.

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6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director of Nuclear Operations or his designee and shall meet or exceed the requirements and recommendations of Section 5.5 of ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees; and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The Plant Review Board shall function to advise the Director of Nuclear Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of the following personnel:

Member:	Technical Support Manager
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Plant Services Manager
Member:	Engineering Manager
Member:	U1, U-2, U-3 ~~Operations~~ Superintendents
Member:	STA Supervisor
Member:	~~Training Manager~~ I and C Superintendent
Member:	Radiation Protection and Chemistry Manager
Member:	Quality Systems/Engineering Manager

The Director of Nuclear Operations shall designate the Chairman and Vice-Chairman in writing.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairman, or his designated alternate.

QUORUM minimum

6.5.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, Vice-Chairman, or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The PRB shall be responsible for:

g. a. Review of ~~(1)~~ all Administrative control procedures ~~required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Director of Nuclear Operations, or designated alternate that affect Nuclear safety.~~ and changes

~~b. Review of all proposed tests and experiments that affect nuclear safety.~~

h. c. Review of all proposed changes to Appendix "A" Technical Specifications.

~~d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.~~

a. e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the ~~Director of Nuclear Operations, or designated alternate and to the~~ Nuclear Safety Group (NSG).

b. f. Review of ~~all REPORTABLE~~ EVENTS requiring 24-hour written notification to the Commission.

c. g. Review of unit operations to detect potential nuclear safety hazards.

d. h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Director of Nuclear Operations. ~~or designated alternate or the NSG.~~

~~i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Director of Nuclear Operations with copies to the NSG.~~

~~j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Director of Nuclear Operations with copies to the NSG.~~

f. k. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

RESPONSIBILITIES (Continued)

e. ~~l.~~ Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

~~m. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Director of Nuclear Operations and to the NSG.~~

~~n. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and radwaste treatment systems.~~

AUTHORITY

6.5.1.7 The PRB shall:

~~a. Recommend in writing to the Director of Nuclear Operations, or his designated alternate approval or disapproval of items considered under Specification 6.5.1.6a. through d. above.~~

a. ~~b.~~ Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. ~~through e.~~ above constitutes an unreviewed safety question.

DNO b. ~~c.~~ Provide written notification within 24 hours to the Vice President-Nuclear Production and ~~the Supervisor of the~~ NSG of disagreement between the PRB and the Director of Nuclear Operations; however, the Director of Nuclear Operations ~~, or his designated alternate~~ shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to ~~the Vice President Nuclear Production and the Supervisor of~~ the NSG.

COMMITTEES

6.5.1.9 Except for the review of programs and administrative control procedures, the PRB may establish standing committees to conduct reviews and make recommendations regarding the responsibilities identified in Specifications 6.5.1.6a., b., d., i., j., and 6.5.1.7a. and b. The following conditions must be met for such a committee:

a. The committee chairman shall be a PRB member designated in Specification 6.5.1.2;

b. The committee members shall be designated in writing by the PRB chairman;.

Delete

~~RECORDS~~

~~6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Nuclear Safety Group.~~

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.2.1 The Director Nuclear Operations (DNO) shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Manager of Technical Support or the Manger of Engineering as previously designated by the Director of Nuclear Ops. Test results shall be approved by the Director of Nuclear Operations or the Manager of Technical Support.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the DNO; or by the Manager, Technical Support as previously designated by the DNO.

New

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the DNO to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the DNO, the Manager Technical Support, the Manager Operations, or the Manager Maintenance.

6.5.2.6 Review of the station security program, and implementing procedures, and submittal of recommended changes shall be approved by the DNO and transmitted to the Vice President-Nuclear Production and to the NSG.

ACTIVITIES (Continued)

6.5.2.7 Review of the station emergency plan, and implementing procedures, and submittal of recommended changes shall be approved.

6.5.2.8 The DNO shall assure the performance of a review by a qualified individual/organizaiton of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence.

6.5.2.9 The DNO shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

New

~~6.5.3 NUCLEAR SAFETY GROUP (NSG)~~

FUNCTION

6.5.3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Quality assurance practices

COMPOSITION

6.5.3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

~~COMMITTEES (Continued)~~

~~c. The responsibility, authority, and functions of the standing committees, including such matters as quorum requirements and documentation of committee activities shall be defined in administrative control procedures;~~

~~d. A report of the activities of each committee shall be made to the PRB at least once per calendar month; and~~

~~e. Any matters which cannot be resolved among committee members, or which the committee feels may involve a change to the technical specifications or an unreviewed safety question shall be forwarded to the PRB for review prior to implementation.~~

Delete

6.5.~~2~~3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.~~2~~3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices

COMPOSITION

6.5.~~2~~3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.~~2~~3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

program and its implementation

REVIEW

6.5.~~2~~3.4 The NSG shall be responsible for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question; This review will be done by the audit of selected safety evaluations.

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REVIEW (Continued)

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS requiring 24-hour notification;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the NSG or the Vice President-Nuclear Production.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

prepared monthly for the Manager, Nuclear Safety, who will

AUDITS (Continued)

~~i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.~~

~~j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.~~

~~k.~~ i. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.~~2~~3.6 The NSG shall report to and advise the Manager of Nuclear Safety on those areas of responsibility specified in Specifications 6.5.~~2~~3.4 and 6.5.~~2~~3.5.

RECORDS

6.5.~~2~~3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be distributed ~~monthly~~ to the Vice President-Nuclear Production, Director of Nuclear Operations~~, Manager of Nuclear Safety,~~ and to the management positions responsible for the areas audited.

6.6 REPORTABLE EVENT-ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS: Requiring 24 hour notification

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted to the ~~Supervisor of the NSG~~ Manager of Nuclear Safety and the Vice President-Nuclear ~~Production~~

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production~~, the Supervisor of the NSG,~~ and Director of Nuclear Operations and Manager Nuclear Safety shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the Supervisor of the NSG and the Vice President-Nuclear Production within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC Addressable Constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5.1 and approved prior to implementation. Programs and administrative control procedures shall be approved by the Director of Nuclear Operations, or designated alternate. Implementing procedures shall be approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

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PROCEDURES AND PROGRAMS (Continued)

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant ~~management~~ supervisory staff, at least one of whom ~~holds a Senior Reactor Operator's License~~ is the Shift Supervisor or Assistant Shift Supervisor on the unit affected.

c. The change is documented, reviewed in accordance with Specification 6.5.1 and approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the NSG at least once per 24 months:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, ~~the post-accident sample return piping of the radioactive waste gas system~~, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

Will be written and issued prior to system being declared operational

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(1) Training of personnel, and

(2) Procedures for monitoring.

e. Post-accident Sampling

will be written and approved prior to plant being declared operational

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencment of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,** e.g., reactor operations

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

ANNUAL REPORTS (Continued)

and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps**

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive ~~liquid and~~ gaseous effluents released from the unit or station during the previous calendar year. ~~This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period.~~ All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. ~~The meteorological conditions concurrent with the time of release of radioactive materials in gaseous~~

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

~~effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. [For ORs: approximate and conservative approximate methods are acceptable.]~~ The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from ~~liquid and~~ gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous ~~and liquid~~ radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.

f. Records of reactor tests and experiments.

RECORD RETENTION (Continued)

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of PRB meetings ~~and of NSC activities~~.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

m. Records of audits performed under the requirements of Specifications 6.5.2.8 and 6.8.4.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

*Radiation Protection personnel or personnel escorted by Radiation Protection personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Radiation Protection Supervisor or his designated alternate in the REP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area. ~~and the maximum allowable stay time for individuals in that area.~~ For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems*, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information;

*Measurement made at 18 inches from source of radioactivity.

PROOF & REVIEW COPY

PROCESS CONTROL PROGRAM (Continued)

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in ~~liquid and~~ gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto; X

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in ~~liquid and~~ gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; X

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

TABLE 4.11-2 (Continued)

TABLE NOTATION

[b]Analyses shall also be performed following SHUTDOWN, STARTUP, or a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.

[c]Sampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

[d]Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

[e]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each SHUTDOWN, STARTUP, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

[f]Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

[g]The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

[h]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Radioactive Effluent Release Report.

RAS 9/12/84

Provide the code letters in parentheses which define sample locations on the maps and tables of the ODCM.

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TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	SAMPLING AND COLLECTION FREQUENCY[a]	TYPE AND FREQUENCY OF ANALYSIS	NUMBER AND APPROXIMATE LOCATION OF SAMPLES[a]
Airborne			
Radioiodine and particulates	Continuous sampling collected weekly[c] or more frequently if required dust loading	Gross beta weekly; I-131 weekly; gamma spectrum monthly; composite of filters[d,e]	Samples from 5 locations: 3 samples at or near the SITE BOUNDARIES, in different sectors of the highest calculated annual average ground level D/Q.* X 1 sample from areas of special interest, which is from the vicinity of a community having the highest calculated annual average D/Q. 1 sample from a control location 15-30 km (10-20 mi) distant and in the least prevalent wind direction.
Direct radiation[b]	Quarterly	Gamma dose quarterly	40 stations with two or more dosimeters for measuring dose rate continuously, placed as follows: an inner ring of stations at the site boundary and an outer ring in the 4-to-5 mi range from the site with a station in each sector of each ring, except the WNW sector, which is inaccessible (16 sectors x 2 rings minus 1 = 31 stations). 7 additional stations are in local schools and population centers; 2 other stations are used as controls.

*D/Q refers to average annual relative ground deposition rate.

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	SAMPLING AND COLLECTION FREQUENCY[a]	TYPE AND FREQUENCY OF ANALYSIS	NUMBER AND APPROXIMATE LOCATION OF SAMPLES[a]
Waterborne			
Surface	Monthly composite of weekly grab sample	Gamma spectrum monthly; tritium quarterly	Water storage reservoir evaporation pond
Ground	Quarterly grab sample	Tritium and gamma spectrums quarterly	2 onsite wells[g]
Drinking (well)	Monthly composite of weekly grab sample	Gross beta and gamma spectrums monthly; tritium quarterly[h]	3 wells from surrounding residences
Ingestion			
Milk	Semimonthly for. animals on pasture; other-wise, monthly	Gamma spectrum and radioiodine semi-monthly (when animals are on pasture) or monthly[h] at other times.	Local dairy
Food products	At harvest[i]	Gamma spectrum and radioiodine monthly[h]	Local farms, one sample of each principal class of food products from any area that is irrigated by water in which plant liquid wastes have been discharged.

TABLE 4.12-1 (Continued)

TABLE NOTATION

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

[c]LLD for drinking water samples.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the methodology and parameters in the ODCM shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, Director of Nuclear Operations, and the Supervisor, Nuclear Safety Group (NSG).

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager of Nuclear Safety and Supervisor of the NSG.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff and shall meet or exceed the minimum qualifications of ANS 3.1-1978 ~~as endorsed by~~ Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents. ×

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*Not responsible for sign-off function.

AIEIS

ARIZONA NUCLEAR POWER PROJECT
Post Office Box 21666 Phoenix. Arizona 85036

received 9/19/84

JND

September 14, 1984
ANPP-30516

Director of Nuclear Reactor Regulation
Attention: Mr. George Knighton, Chief
Licensing Branch No. 3, Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: PVNGS Units 1, 2, and 3
Proof and Review Technical Specifications
Docket Nos. STN 50-528/529/530
File: 005-419.05

Dear Mr. Knighton:

Your letter dated August 14, 1984 transmitted to APS our copy of the PVNGS proof/review Technical Specifications. Your letter requested us to review and respond to our proof and review technical specifications. Due to the detail of our review we are submitting our response to you one day late. This was discussed with M. Licitria, D. Brinkman (NRC) and S. R. Frost (APS). The one day did not present any problems for the reviewers.

In performing our PVNGS Technical Specification Review we developed a committee to review and comment on the NUREG 0212 Rev. 3 approximately two years ago. Our committee consisted of offsite engineering, Licensing, onsite Operations, H.P./Chemistry, Maintenance, Engineering, Startup, QA, STA/ISEG, I and C, Training, Bechtel Engineering and Combustion Engineering. This committee worked closely with the NRC reviewer to develop a set of technical specifications that represented PVNGS.

This committee functioned, as follows, to mold the CE Standard Tech Specs so they would not only represent the design of PVNGS but also represent how the plant will be operated:

1) Utilize our own Plant Specific experience to review systems, their functions, parameters and system names.

2) Discussed Tech Spec problems with operating units throughout the industry.

3) Held review meetings with various operating units.

4) Had operating experienced units review/comment on our proof/review tech specs.

5) We have used our Tech Specs during our startup program to see if we can live with the various specs and associated equipment in order to eliminate future problems (i.e., pump performance etc.).

6) Monitored Federal Register to see if any Tech Spec changes other plants obtained would applying to PVNGS.

7) Reviewed various operating experiences (i.e., LERs, some inspection reports, etc.) to see if they could affect the Tech Specs.

8410020130

8) Compared the Tech Specs to the PVNGS FSAR for consistency.

9) Compared the Tech Specs to the PVNGS SER for consistency.

10) Compared the Tech Specs to the CESSAR FSAR for consistency.

11) Compared the Tech Specs to the CE-SER for consistency.

12) We have used our vendor's experience and support from the beginning to develop our Tech Spec.

13) We have had our DCPs reviewed to see if Tech Spec changes are needed.

14) We have trained our operators in our "marked up" Tech Specs over the past 2 years.

15) We have utilized our Tech Specs on the PVNGS Plant Specific Simulator.

16) We have monitored/solicited questions and interpretation problems from Training and Operations and revised our Tech Specs to make the Tech Specs clear for everyone.

17) We have written our procedures from our marked up Tech Specs and as problems arise we may have changed the spec.

18) Continuous discussions over the past two years with our resident inspectors and resolving their problems either through discussion or revision to the Tech Spec.

We believe that we have conducted a detailed review of the PVNGS Tech Specs and have a good operational document if issued in a final form as we have amended Attachment A. All of our changes marked in the proof/review copy have justifications in Attachment B. Many of the changes that are identified in this marked up proof/review copy have been submitted along with their justifications over the past years.

We feel very strongly that we need all of the attached changes for the following reasons:

1) This is how we will operate the unit.

2) Some of the changes are a "human factors" consideration that will hopefully eliminate errors that other operating plants have experienced.

3) To avoid massive amount of Tech Spec changes after we go operational (as experienced by other utilities).

The new NRC Tech Spec program requires that the licensee certify their Tech Specs prior to final acceptance. It is our position that in order to certify the Tech Specs that they not only have to reflect the design of PVNGS but also its operation. Therefore, we will need to implement all the changes identified in Attachment A to this letter.

If you have any questions please contact S. R. Frost (602) 943-7200, extension 6183.

Very truly yours,

E E Van Brunt / TBK

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Production
ANPP Project Director

EEVB/SRF/wpc

cc: E. Licitra (w/a)
J. B. Martin (w/a)
R. Zimmerman (w/a)
G. Fiorelli (w/a)

Accepted jrd

me

PG 3/4 3-24 Typo.

PG 3/4 3-25 DELETE:

Old page. Insert new table.

RSB ICSB

JUSTIFICATION:

The new table is a more useable table to the operators. It breaks the systems down to where everyone understands. The old table was confusing in that the times represented showed the TOTAL time of the System response times not done of the key subsystem response times such as shown in the new table.

PG 3/4 3-26 DELETE:

Note. See page.

RSB ICSB

JUSTIFICATION:

The response times for the AFWP, are shown in the body of the table. Showing a note with another response time doesn't match is not really necessary. We have had alot of questions generated as to why the note if the AFWP, response times are in the Body of the table. Operations are continuously confused by both values; they want only 1 to worry about.

PG 3/4 3-27, 28, 29, 30, 31 Change automatic actuation logic channel functional test to R

RSB ICSB

JUSTIFICATION:

PVNGS will be submitted a letter to the NRC next week providing detailed justification. In summary we do not want to test this logic at power operation because various equipment would actuate and could trip the unit. Many other utilities are having problems meeting this Tech. Spec. and are going to be asking for a Tech. Spec change, in fact one set in there proposed change and justification about a month ago.

PG 3/4 3-34, 35 CHANGE:

Table 3.3-6. See page.

METB

JUSTIFICATION:

Incorporation of RETS ("Radiological Effluent Technical Specifications") into the STS ("Standard Technical Specifications") has resulted in duplication of operability requirements for the Effluent Monitoring System.

Inconsistency in applicability and discrepancies in reporting requirements further complicate this situation. Precedent has been set in the NRC Region II for the resolution of this problem.

Technical Specification requirements for the effluent monitors have been deleted from Table 3.3-6, page 3/4, 3-35 as Table 3.13 already addresses the required parameters. Applicability requirements of Table 3.3-6 have been retained in Table 3.3-13.

Deletion of Item C - Ru 148 and 149 is based on the fact that these are post-accident monitors and will only serve a purpose in the event of an accident. These monitors were placed in this table by APS and further evaluation shows that they should be deleted. RU-1 (Item 2.A) is the monitor that will be used to monitor containment atmosphere during operation conditions. Action Item 24 needs to be deleted in this section based on that it is not any longer used here but its incorporated in Action Statement 37.

PG 3/4 3-37, 38 and 63

<u>CHANGE:</u>

See Page.

RETB

<u>JUSTIFICATION:</u>

In order to fully complete the incorporation of the RETS into the STS, the surveillance requirements of Table 4.3-3 and 4.3-8 have been revised to reflect the above changes.

1. Action Statement 37 has been changed to allow for the implementation of the preplanned alternate sampling program allowed for effluent monitors under the former Action Statement 27 of Table 3.3-6. This program alleviates the rigorous sampling frequency when operational source terms justify sampling at a lower frequency, i.e., prior to initial criticality, prior to exceeding 5% power, or during Modes 5 and 6.

2. Action Statement 38 should be deleted as it is not referenced in the Tables as they appear in the "Proof and Review" copy.

3. Action Statement 39 has been revised to delete the sentence "With both channels inoperable be in at least Hot Standby within 6 hours."

   If there is condition of explosive concentrations of gas in the gaseous radwaste system, the reduction of

power level in the reactor (decrease in coolant temperature and pressure) may aggravate the problem by inducing degassification of the primary coolant.

We are taking fuel building Ventilation exhaust and condenser vacuum plump/gland seal monitor out of Table 4/3-3 because they are duplicated in Table 3.3-13.

PG 3/4 3-39 CHANGE:

LPB

Surveillance 4.3.3.2b ~~and 2~~

4.3.3.2.a not acceptable because it would allow not testing just after incore detectors returned to service because they were tested 7 days earlier

JUSTIFICATION:

By adding the "7 days or more have elapsed since the last use" provides a time period for guidance for a channel check. This addition is consistent with SONGS tech. specs. They said without this clarification that there were problems with the region and plant interputation.

Delete last sentence of 4.3.3.2.b. We have no means to perform the calibration of the incore detectors. This is a function being performed by all suppliers of incores. Other utilities that have this Tech. Spec. say they cannot really meet this Spec. See justification for Page 3-64-67 on Page 15 for *** justification. —

PG 3/4 3-41 CHANGE:

SGEB

3.a (setpoint 0.02g).

JUSTIFICATION:

FSAR Section 3.7.4.3, pg. 3.7-32.

PG 3/4 3-44 CHANGE:

1a and 1b.

MEB

JUSTIFICATION:

NUS recently supplied information showed the instrument range to be:

1 to 50 mph
1 to 50 mps

PG 3/4 3-52 CHANGE:

The LCO and Applicability. See page.

CEB

add "Fire" to FPER "Fire" Zone in Table 3.3-11

PROOF

JUSTIFICATION:

PVNGS does not use "fire detection zone" terminology our fire detection equipment is located in the fire zone.

Change from Function A(B) to Function X(Y) is to be consistent with Table 3.3-11.

PG 3/4 3-56 CHANGE:

CEB

Note Typo's.

JUSTIFICATION:

Typo's.

PG 3/4 3-58 CHANGE:

Typo's, see page. Accepted JMW

~~CPB~~ ~~ICSB~~

PG 3/4 3-60, 61, 62. CHANGE:

See pages.

~~DSB~~ METB

JUSTIFICATION:

instead of * - this monitor is only needed during waste gas release; therefore, requiring this instrument to be operable all the time is not appropriate.

is used because this monitor is only needed during the conditions cited in the new note. There is no radiological hazard except only in these conditions.

#RU-144 has been deleted from Table 3.3-13 as the accident range noble gas monitors are required under commitments to NUREG-0737. The presence of RU-144 in this location causes confusion of the interpretation of the operability requirements of RU-143 and RU-144.

The present configuration of the Technical Specification Table 3.3-13 would lead one to believe that RU-144 would be adequate to replace the function of RU-143. This is not an appropriate interpretation. RU-144 is not in any way a redundant channel to RU-143 which is the normal range noble gas effluent monitor.

Note change Justification were identified above.

PG 3/4 3-63 CHANGES NOTED ON PAGE 13 OF THIS ATTACHMENT

~~DSB~~ METB

PROOF AND

PG 3/4 3-64, 65, 66 67

CHANGES:

Add footnotes.

~~MEB~~
METB

JUSTIFICATION:

These notes are added identifying how PVNGS will operate.

Modify functional test requirement for item 1.a and 1.b to P###. Add footnote: Functional test shall consist of, but not be limited to, a verification of system isolation capability by insertion of a simulated alarm condition.

Complete system functional testing is accomplished on a quarterly basis. The depth of this functional testing is beyond the scope of verifying system operability prior to commencing a purge/release.

PG 3/4 3-68

DELETE:

ASB

Tech. Spec.

JUSTIFICATION:

We have provided vast amounts of Justification to delete this Tech. Spec. This Tech. Spec. is to protect against turbine missiles. We have shown that in the event we do have a turbine missile that it would not hit any safety related equipment, containment or the other units. Our containment building is perpendicular to the turbine not parallel as other nuclear plant. One nuclear plant got this Tech. Spec. deleted for the same reasons. Therefore, it is our belief that this Tech. Spec. does not serve a significant purpose.

PG 3/4 4-~~19~~ 18
4-20

CHANGE:

RSB
ASB
MEB

Tech. Spec. See page.

JUSTIFICATION:

This is a Tech. Spec. we committed to Catauba Nuclear Station in our letter to the NRC ANPP-30290, dated August 21, 1984.

PG 3/4 4-27 Typo.

PG 3/4 4-29 NEW TABLE WILL BE SUPPLIED IN A WEEK

MEB/
RSB
MEB

PG 3/4 4-26

METB

a more detailed action statement to instruct the operators as to what they need to do in situations not identified in the old action statement.

PG 3/4 6-37 CHANGE:

Surveillance 4.6.4.2.b.1. Delete all.

JUSTIFICATION:

The word "ALL" needs to be deleted. There is some instrumentation that is not "vital" or needed to insure proper operation of the recombiner to-perform its intended safety functions.

PG 3/4 6-38 CHANGE: ICSB METB

LCO Delete the last six words of the LCO ... "In each of the three units."

JUSTIFICATION:

This statement is confusing in its original form. Our people interpret it to mean that power needs to be supplied from all three units no matter where the containment hydrogen purge cleanup system is located. The proposed change clarifies this problem.

PG 3/4 6-39 ADD: METB

Page. This page was deleted in the Proof/Revision copy of the Tech. Spec.

PG 3/4 7-2,3 CHANGE: ASB

See page.

JUSTIFICATION

CE supplied input.

PG 3/4 7-4 7-5. CHANGE: ASB

Surveillance Requirements 4.7.1.2.a.1, 4.7.1.2.a.3, and 4.7.1.2.c. See pages.

JUSTIFICATION:

Delete the last sentence in 4.7.1.2.1.1. This statement will be broken out into item 4.7.1.2.d. This is done to add clarification.

PG3/4 7-1 mo

PG 3/4 9-12 DELETE:

me

See page.

JUSTIFICATION:

Typo.

PG 3/4 9-13 CHANGE:

ASB
AEB

LCO. See page.

JUSTIFICATION:

The correct number (22 feet 8 inches) of water shall be maintained over the top of the storage racks is the correct LCO. The 22 feet 8 inches is needed to ensure the minimum water depth to remove a nominal 99% of the assumed gap activity released from a ruptured irrigated fuel assembly lying on its side on top of the storage racks.

PG 3/4 10-6 CHANGE:

RSB
we

LCO Item C. See page.

JUSTIFICATION:

We want to add "Key-Locked" to LCO Item C. This depicts the actual way we will operate and maintain the valves of the Safety Injection tanks to be open.

PG 3/4 10-8 CHANGE:

RSB

LCO Item b. See page.

JUSTIFICATION:

The addition of ..."or not to go below 254 psig" is needed to alert the operator of this operating limit for this special test exception.

PG 3/4 11-2 CHANGE:

RAB
~~MEB~~
METB

Table 4.11-1. See page.

JUSTIFICATION:

Number left off Table.

RAB
METB

PG 3/4 11-3 CHANGE:

Footnote b. See page.

PROOF AND REVIEW

JUSTIFICATION:

METB

PVNGS does not have a method to thoroughly mix liquid wastes the way it is presently stated in the Proof/Revision Tech. Spec.

Footnote b was rewritten to comply with the way PVNGS will perform this function.

PG 3/4 11-5 DELETE:

RAB

Tech. Spec. Request the deletion of this specification, its basis B 3/4 11.1.2, and removal of reference to liquid radioactive effluent releases with paragraph 2, 3, 6, and 8 Technical Specification 6.9.1.8.

JUSTIFICATION:

There are no liquid release points from which activity is discharged for use by the general public. In addition, sampling required by Technical specification Table 4.11-1 will monitor liquid discharges to the evaporation pond. When this Tech spec (3/4 11.1.2) was initially requested to be changed the reviewer developed Tech. Spec. 4.11-1 for our evaporation pond so 3/4 11.1.2 could be deleted since we are a zero release plant.

PG 3/4 11-6 CHANGE:

LCO. See page.

METB

JUSTIFICATION:

BPC recommended we change the 10 Ci to 500 Ci in this Tech Spec. This change is supported by the PVNGS FSAR, Section 2.4-13 (page 2.4-118). For this accident BPC assumed a 100 Ci event multiplied this number by 5 and you still are below the required MPC limit; therefore, the change to 500 Ci is acceptable and supported by the FSAR and will not exceed the MPC allowable limit.

11-7
11-8
PG 3/4 11-10 CHANGE:

RAB
METB

Footnote b. See page.

JUSTIFICATION:

PWR's do not vent off gas directly to atmosphere as do BWR's. Any increase in gaseous activity due to power level change would be the result of increases in the fuel leakage rate to the primary coolant. Primary system valve leakage to containment atmosphere or to the auxiliary building atmosphere would be the sources of

PG 3/4 11-14

METB

METB

discharge to the environment. Both of these discharge pathways pass through the plant vent. An increase in the noble gas monitor count rate of greater than a factor of 3 would indicate that the isotopic mixture of the effluent had change sufficiently to warrant a re-analysis of alarm setpoints and projected offsite dose. Grab samples of noble gas and Tritium are analyzed weekly for the generation of Gaseous Effluent Release Permits. It is felt that the weekly sampling frequency in addition to the Containment Atmosphere Monitor readings will monitor plant vent discharges so they will remain below Technical Specification and 10CFR20 limits. The Fuel Building Exhaust discharge should not be affected by power level changes.

PG 3/4 11-15 CHANGE: METB

Surveillance 4.11.2.6. See page.

JUSTIFICATION:

Add ... "During Addition" to the Surveillance Requirement. This is needed so it identifies how we will perform this surveillance more accurately thus, avoiding possible errors. Since this spec is for quantity of radioactivity in the gas storage tank, it should only be monitored during additions to that tank.

PG 3/4 11-17 CHANGE:

LCO. See page. METB

JUSTIFICATION:

PVNGS does not have filter sludges; therefore, this reference in the LCO is not applicable to the PVNGS design.

PG 3/4 11-18 CHANGE: RAB

Surveillance 4.11.4.1. See page.

JUSTIFICATION:

Delete all references to liquid effluent for the justification stated above for 3.11.1.2.

PG 3/4 5-6 CHANGE: ASB ~~ETS~~

5.6.1.1.b.

~~B3/4 1-1~~ ~~ETS~~

PG 5-7 5-8 CHANGE:

See page.

JUSTIFICATION

CE provided input concerning plant cycles and transients for PVNGS.

RSB. CPB

CH6 CHANGE:

See page.

JUSTIFICATION

Chapter 6 was rewritten to reflect the administrative process in which PVNGS operates. These changes have previously been accepted by Region V at other operating plants.

METB See next page

BASIS CHANGES

See pages.

JUSTIFICATION

The basis has been revised in pacts to more accurately reflect our plant.

See next page

pages 6-13 to 6-25 as appropriate

Bases

Pg	Branch(s)
B 3/4 1-1	CPB
B 3/4 1-2	CPB
1-3	CPB
2-1	CPB
3-2	METB
3-3	CEB
3-4	ASB
4-1	RSB
4-6	RSB MtEB
4-7	RSB MtEB
4-11	RSB MtEB
5-1	CPB
5-3	RSB
6-2	CSB
6-3/4	AEB CSB

Pg	Branch(s)
B3/4 7-1	RSB ASB
7-2	RSB ASB
7-3	ASB
7-4	MEB
7-6/7	CEB
9-3	AEB ASB
10-1	CPB
10-2	RSB
11-1/2	RAB METB
11-5	METB

Justification from applicant for the following proposed changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications:

Table 3.3 - 13, Page 3/4 3-63	METB
Bases Section 3/4 3.3.1, Page 3 - 2	METB
3/4.11.2.1, Page 3/4 11 - 7	RAB/METB
Table 4.11-2, Page 3/4 11-10	RAB/METB
Table 4.4-4, Page 3/4 4-26	METB
3.11.2.5.b, Page 3/4 11-14	METB

MTEB → Table 3.3-13 Page 3/4 3-63

1. Applicability designators * and ** along with their descriptions must be left in as they refer to the Plant Vent Monitor and the Explosive Gas Monitoring System respectively.

2. Action 37 must be changed to read the following:

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days, provided the actions of "A" or "B" are performed.

A. Initiate the preplanned alternate method of sampling the appropriate parameter(s).

B. Take grab samples at least once per 12 hours and analyze for gross activity within 24 hours.

Current Justification

Justification: This allows for ~~the~~ the implementation of the preplanned alternate sampling program allowed for effluent monitors under the former Action Statement 27 of Table 3.3-6

Action Statement 37 as amended at Proof and Review Approval session is unacceptable as it imposes a ~~Routine~~ 14 Reporting Requirement where it was only

necessary to explain inoperability of monitors in the Semi Annual Effluent 1.21 Report. (prior to meeting)

Monitoring for effluent noble gas concentrations may be accomplished by grab sampling at a specified frequency. Monitor installation should not be a requirement when a monitor is inoperable.

There are no process monitors listed in this Table this part of the Action 37 refers to the Containment Atmosphere monitor and should not be listed here as it is in Table 3.3-6 Action 27

MTEB ~~###~~ Bases under Section 3/4 3.3.1 Page B3/4

B 3/4 3-2

Delete (3)! We are committed to requirements of NUREG 737 and Reg Guide 1.97 already no need for this in Tech. Specs. Already in 6.8.1.a under Admin Controls

RAB
MTEB

3/4.11.2.1 Page 3/4.11-7

Change to:

6. For [Iodine-131, Iodine 133,] Tritium and for all radio....

Justification: These are only isotopes of Iodine of consequence to off-site dose due to gaseous effluent releases as described in NUREG 017 others would be of consequence perhaps in liquid release of be considered in Solidified Waste Part 61 requirements. This will allow agreement with Tech Spec 3.11.2.3

3/4 11-7

4.11.2.1.2

Change to:

The Dose Rate due to Iodine-131, Iodine 133, Tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents... ⇒

RAB
MTEB

Table 4.11-2 Page 3/4 11-10

3/4 11-10

Justification

Delete item "d" as this is not referenced in the Table and sampling frequency is ridiculous. This parameter is ~~the~~ sampled at the Plant vent monthly ~~which~~ This should satisfy reporting requirements as tritium equilibrium in the refueling water and coolant would not vary significantly over this period of time

Frequency of Fuel Building Tritium sampling item "f" should be monthly also for reason given above.

Admin Control
6.8.1.j
add: "Preplanned Alternate Monitoring and Sampling Program" to list under PRB control.
This implements alternatives ~~changes~~ to monitor and sampling scedules in Table 3.3-6 and Table 3.3-13

METB Table 4.4-4 Page 3/4 4-26

✓ Add qualifier to 4.6 ** "One sample is sufficient if plant has ~~come to a~~ gone through SCRAM condition or if transient is complete in 6 hours".

Justification: Interpretation is possible that more than one sample is required if power level varies greater than 15%

Table 4.11-2 Page 3/4 11-10

-change to

item 6. ... If noble gas monitor is inoperable, samples must be obtained As soon as possible and must be analysed within a Four hour period. This requirement does not apply to the Fuel Building ~~Vent~~ Exhaust

METB 3.11.2.5.b Page 3/4 11-14

3/4 11-14

change "within 1 hour" to "within 6 ho[urs]"

Justification: Release rate is limited t[o] 50 cfm and average release of tank will take 6 hours

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3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
A. Fuel Pool Area RU-31	1	**	≤5mR/hr	10^{-1} to 10^{4}mR/hr	22 & 24
B. New Fuel Area RU-19	1	*	≤5mR/hr	10^{-1} to 10^{4}mR/hr	22
C. Containment RU-148 RU-149	2	1,2,3,4	≤2 R/hr	1R/hr to 10^{7}R/hr	27
D. Containment Power Access Purge Exhaust RU-37 & RU-38	1	#	≤2.5mR/hr	10^{-1} to 10^{4}mR/hr	25
E. Main Steam					
1) RU-139 A&B	1	1,2,3,4	≤10mR/hr	10^{-3} to 10^{4}R/hr	27
2) RU-140 A&B	1	1,2,3,4	≤10mR/hr	10^{-3} to 10^{4}R/hr	27
2. Process Monitors					
A. Containment Building Atmosphere RU-1	2	1,2,3,4			23 & 27
1) Particulate			≤2.3×10^{-6}μCi/cc Cs-137	10^{-9} to 10^{-4}μCi/cc	
2) Gaseous			≤6.6×10^{-2}μCi/cc Xe-133	10^{-6} to 10^{-1}μCi/cc	
B. Noble Gas Monitors					
1) Control Room Ventilation Intake RU-29 & RU-30	1	All MODES	≤2×10^{-6}μCi/cc	10^{-7} to 10^{-1}μCi/cc	26

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2) Fuel Building Ventilation Exhaust RU-145	1	**	Per ODCM	10^{-7} to 10^{-1} µCi/cc	27 & 24
3) Condenser Vacuum Pump/ Gland Seal Exhaust RU-141	1	1,2,3,4	Per ODCM	10^{-7} to 10^{-1} µCi/cc	27
4) Plant Vent Gaseous RU-143	1	All MODES	Per ODCM	10^{-7} to 10^{-1} µCi/cc	27
5) Waste Gas Decay Tank Discharge RU-12	1	##	Per ODCM	10^{-3} to 10^{2} µCi/cc	***

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
~~***ACTION in accordance with Table 3.3-23 ACTION 35.~~
#When purge is being used.
~~##During waste gas release.~~

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TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.4.5.1).

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.12).

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.9).

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1. Complete the actions of "A" or "B".

   A. Initiate the preplanned alternate method of monitoring the appropriate parameter(s).

   B. For process monitors, place moveable air monitor in-line or take grab sample at least once per 24 hours.

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
A. Fuel Pool Area RU-31	S	R	M	**
B. New Fuel Area RU-19	S	R	M	*
C. Containment Power Access Purge Exhaust RU-37 & RU-38	P#	R	P###,W##	##
D. Containment RU-148 & RU-149	S	R	M	1,2,3,4
E. Main Steam RU-139 A&B RU-140 A&B	S	R	M	1,2,3,4
2. Process Monitors				
A. Containment Building Atmosphere RU-1				
1) Particulate	S	R	M	1,2,3,4
2) Gaseous	S	R	M	1,2,3,4
B. Noble Gas Monitors				
1) Control Room Ventilation Intake RU-29 & RU-30	S	R	M	All MODES
2) Fuel Building Ventilation Exhaust RU-145	S	R	M	**
3) Condenser Vacuum Pump/Gland Seal Exhaust RU-141	S	R	M	1,2,3,4

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TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4) Plant Vent RU-143	S	R	M	ALL MODES
5) Waste Gas Tanks Discharge RU-12	P	R	P	###

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*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
#If purge is in service for greater than 12 hours, perform once per 12-hour period.
##When purge system is in operation.
~~###During waste gas release.~~
THE FUNCTIONAL TEST SHOULD CONSIST OF, BUT NOT LIMITED TO A VERIFICATION OF SYSTEM ISOLATION CAPABILITY BY THE INSERTION OF A SIMULATED ALARM CONDITION

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METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

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OPERATIONS
LICENSING

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT		LOCATION	MINIMUM OPERABLE
1.	WIND SPEED		
	a. 1Ø to 50 mph,	Nominal Elev. 35 feet	1
	b. 1Ø to 50 mph,	Nominal Elev. 200 feet	1
2.	WIND DIRECTION		
	a. 0°-360°-180°,	Nominal Elev. 35 feet	1
	b. 0°-360°-180°,	Nominal Elev. 200 feet	1
3.	AIR TEMPERATURE - DELTA T		
	a. -6°F to 6°F,	Nominal Elev. 35 feet-200 feet	1

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OPERATIONS
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TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 35 feet	D	SA
b. Nominal Elev. 200 feet	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 35 feet	D	SA
b. Nominal Elev. 200 feet	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 35 feet - 200 feet	D	SA

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-13.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel; or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to OPERABLE status within the time specified in the ACTION or explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8, why this inoperability was not corrected within the time specified.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-8.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GASEOUS RADWASTE SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release #RU-12	1	~~*~~ #	35
b. Flow Rate Monitor	1	~~*~~ #	36
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	1	**	39
b. Oxygen Monitor	1	**	39

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NUCLEAR OPERATIONS LICENSING

TABLE 3.3-13 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3.	CONDENSER EVACUATION SYSTEM			
	a. Noble Gas Activity Monitor #RU-141	1	1, 2, 3, 4	37
	b. Iodine Sampler	1	1, 2, 3, 4	40
	c. Particulate Sampler	1	1, 2, 3, 4	40
	d. Flow Rate Monitor	1	1, 2, 3, 4	36
	e. Sampler Flow Rate Measuring Device	1	1, 2, 3, 4	36
4.	PLANT VENT SYSTEM			
	a. Noble Gas Activity Monitor #RU-143 and #RU-144	1	*	37
	b. Iodine Sampler	1	*	40
	c. Particulate Sampler	1	*	40
	d. Flow Rate Monitor	1	*	36
	e. Sampler Flow Rate Measuring Device	1	*	36

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5.	FUEL BUILDING VENTILATION SYSTEM			
a.	Noble Gas Activity Monitor #RU-145	1	##	37
b.	Iodine Sampler	1	##	40
c.	Particulate Sampler	1	##	40
d.	Flow Rate Monitor	1	##	36
e.	Sampler Flow Rate Measuring Device	1	##	36

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DURING WASTE GAS RELEASE

WITH IRRIDIATED FUEL IN THE STORAGE POOL, OR IN MODES 1, 2, 3 OR 4

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TABLE 3.3-13 (Continued)

TABLE NOTATION

~~* At all times.~~

STET

~~** During GASEOUS RADWASTE SYSTEM operation.~~

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this GASEOUS RADWASTE SYSTEM may continue for up to 14 days provided grab samples are collected at least once per 8 hours and are analyzed within the following 4 hours for the "on service" gas decay tank. ~~With both channels inoperable, be in at least HOT STANDBY within 6 hours.~~

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

ACTION 37 WITH THE NUMBER OF CHANNELS OPERABLE LESS THAN THE REQUIRED BY THE MINIMUM CHANNELS OPERABLE REQUIREMENT, RELEASES VIA THIS PATHWAY MAY CONTINUE FOR UP TO 30 DAYS, PROVIDED THE ACTIONS OF (A) OR (B) ARE PERFORMED COMPLETE THE ACTIONS OF A OR B

A. INITIATE THE PREPLANNED ALTERNATE METHOD OF SAMPLING THE APPROPRIATE PRAMETER(S)

B. TAKE GRAB SAMPLES AT LEAST ONCE PER 12 HOURS AND ANALYZE FOR GROSS ACTIVITY WITHIN 24 HRS

TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS RADWASTE SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1), P###	~~*~~ ##
b. Flow Rate Monitor	P	N.A.	R	Q, P###	~~*~~ ##
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (sequential)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (sequential)	D	N.A.	Q(5)	M	**

PROOF AND REVIEW

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TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	1, 2, 3, 4
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Flow Rate Monitor	D	N.A.	R	Q	1, 2, 3, 4
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	1, 2, 3, 4
4. PLANT VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

PROOF AND REVIEW

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TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
5. FUEL BUILDING VENTILATION SYSTEM					
a. Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	##
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	##
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	##
d. Flow Rate Monitor	D	N.A.	R	Q	##
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	##

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PROOF AND REVIEW

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TABLE 4.3-8 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

DURING WASTE GAS RELEASE
WITH IRRADIATED FUEL IN THE STORAGE POOL OR MODES 1, 2, 3 OR 4
FUNCTIONAL TEST SHOULD CONSIST OF BUT NOT BE LIMITED TO A VERIFICATION OF SYSTEM ISOLATION CAPABILITY BY THE INSERTION OF A SIMULATED ALARM CONDITION

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AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

*With T_{cold} greater than or equal to 500°F.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

MODES 1, 2, 3, 4, and 5:

d. With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries/gram, perform the sampling and analysis requirements of item 4.(a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE

AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	(a) Once per 4 hours, whenever the specific activity exceeds 1.0 µCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram, and	1#, 2#, 3#, 4#, 5#
	(b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. ONE SAMPLE IS SUFFICIENT IF PLANT HAS GONE THROUGH A SCRAM CONDITION OR IF TRANSIENT IS COMPLETE IN 6 HOURS	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

PROOF AND REVIEW

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PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

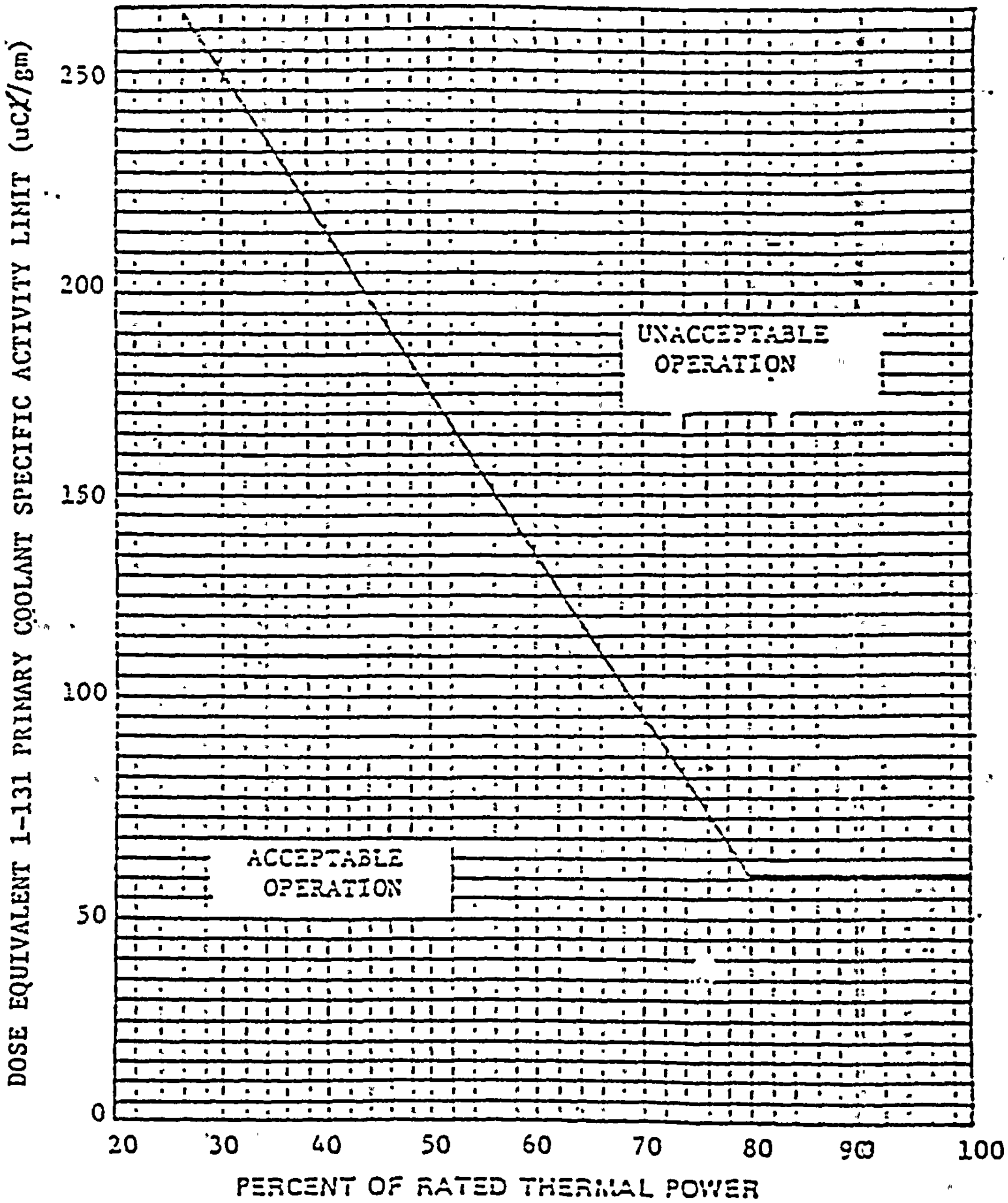

FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 μCi/GRAM DOSE EQUIVALENT I-131

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus. ~~in each of the three units~~.

APPLICABILITY: MODES 1* and 2.*

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm $\pm$ 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

*With less than two hydrogen recombiners OPERABLE.

PROOF AND REVIEW

JUN 1983
RECEIVED
NUCLEAR OPERATIONS LICENSING

3. Verifying a system flow rate of 50 scfm ± 10% during system operation when tested in accordance with ANSI N510-~~1975~~ 1980

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

   PRE FILTERS

   1. Verifying that the pressure drop across the combined 8.4 HEPA filters, and charcoal adsorber banks is less than ~~(8)~~ inches Water Gauge while operating the system at a flow rate of 50 scfm ± 10%.

   2. Verifying that the heaters dissipate AT LEAST 0.5 ~~± 0.1~~ kW when tested in accordance with ANSI N510-~~1975~~ 1980

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99% of the DOP when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of 50 scfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-~~1975~~ 1980 while operating the system at a flow rate of 50 scfm ± 10%.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR OPERATIONS LICENSING

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 SECONDARY SYSTEM LIQUID WASTE DISCHARGES TO ONSITE EVAPORATION PONDS CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material discharged from secondary system liquid waste to the onsite evaporation ponds shall be limited to the lower limit of detectability (LLD) defined as 5×10^{-7} μCi/ml for the principal gamma emitters or 1×10^{-6} μCi/ml for I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

When any secondary system liquid waste discharge pathway concentration determined in accordance with the surveillance requirements given below exceeds the specified LLD, divert that discharge pathway to the liquid radwaste system without delay.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes collected in the chemical waste neutralizer tank shall be sampled and analyzed prior to their batchwise discharge to the onsite evaporation pond in accordance with the sampling and analysis program specified in Table 4.11-1.

4.11.1.1.2 With the concentration of radioactive material in the chemical waste neutralizer tank exceeding the specified LLD, sample and analyze other secondary system discharge pathways in accordance with the sampling and analysis program specified in Table 4.11-1.

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AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

TABLE 4.11-1

SECONDARY SYSTEM LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

SECONDARY SYSTEM LIQUID RELEASE PATHWAY	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/mL)
A. Batch discharges[b]				
1. Chemical Waste Neutralizer Tank	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
2. Steam Generator Blowdown Low TDS Sump*	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
3. Condensate Polishing Low TDS Sump*	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
B. Continuous Releases[d]				
1. Turbine Building Sump*	D Grab Sample	D Grab Sample	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
2. Condenser Area Sumps*	D Grab Sample	D Grab Sample	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}

*Sampling and analysis for pathways 2 and 3 under batch discharges and 1 and 2 under continuous releases are required only when concentration for chemical waste neutralizer tank pathway exceeds the LLD.

PROOF AND REVIEW

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TABLE 4.11-1 (Continued)

TABLE NOTATION

[a]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as microcuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b]A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

[b] PRIOR TO DISCHARGE EACH BATCH SHALL BE ISOLATED AND SAMPLED IN A REPRESENTATIVE MANNER

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

TABLE 4.11-1 (Continued)

TABLE NOTATION

[c]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

[d]A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR OPERATIONS LICENSING

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.3 The quantity of radioactive material contained in each outside temporary tank and the reactor makeup water tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

500

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank or the reactor makeup water tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 The quantity of radioactive material contained in each outside temporary tank and the reactor makeup water tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site (see Figures ~~5.1-1~~ and 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For I-131 AND I-133 ~~all radioiodines~~, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to ~~radioactive materials, other than noble gases~~, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

I-131, I-133, TRITIUM AND ALL RADIONUCLIDES IN PARTICULATE FORM WITH HALF-LIVES GREATER THAN 8 Days.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD) ($\mu Ci/ml$)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[h,g]	$1x10^{-4}$
B. Containment Purge	P Each Purge[b,c] Grab Sample	P Each Purge[b,c]	Principal Gamma Emitters[h,g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. 1. Condenser Vacuum Pump Exhaust 2. Plant Vent 3. Fuel Bldg. Exhaust	M[b,h,e] Grab Sample	M[b]	Principal Gamma Emitters[h,g]	$1x10^{-4}$
			H-3	$1x10^{-6}$
	Continuous[e,f]	W[d,e] Charcoal Sample	I-131	$1x10^{-12}$
			I-133	$1x10^{-10}$
	Continuous[e,f]	W[d,e] Particulate Sample	Principal Gamma Emitters[h,g] (I-131, Others)	$1x10^{-11}$
	Continuous[e,f]	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[e,f]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
D. All Radwaste Types as listed in A., B., and C. above.	Continuous[e,f]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$

PROOF AND REVIEW

RECEIVED AUG 1984 NUCLEAR OPERATIONS LICENSING

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is the smallest concentration of radioactive material in a sample that will yield a net count, above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detectable concentration, as the detection capability for a given instrument procedure and type of sample.

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between the midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:

(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report (ARH-2537 (June 22, 1972).

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

TABLE 4.11-2 (Continued)

TABLE NOTATION

b~~Analyses shall also be performed following SHUTDOWN, STARTUP, or THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1 hour period.~~

cSampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

~~d Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.~~

D ~~e~~ Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). ~~Sampling shall also be performed at least once per 24 hours for at least 7 days following each SHUTDOWN, STARTUP, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.~~ STET

E ~~f~~ Tritium grab samples shall be taken at least MONTHLY ~~once per 7 days~~ from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

F ~~g~~ The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

G ~~h~~ The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

b ANALYSES SHALL ALSO BE PERFORMED FOLLOWING SHUTDOWN, STARTUP, OR A THERMAL POWER CHANGE EXCEEDING 15% OF THE RATED THERMAL POWER WITHIN A 1 HOUR PERIOD UNLESS 1) ANALYSIS SHOWS THAT THE DOSE EQUIVALENT I-131 CONCENTRATION IN THE PRIMARY COOLANT HAS NOT INCREASED MORE THAN A FACTOR OF 3; AND (2) THE NOBLE GAS ACTIVITY MONITOR ON THE PLANT VENT SHOWS THAT EFFLUENT ACTIVITY HAS NOT INCREASED BY MORE THAN A FACTOR OF 3. IF THE ASSOCIATED NOBLE GAS VENT MONITOR IS INOPERABLE, SAMPLES MUST BE OBTAINED AS SOON AS POSSIBLE, ANALYSES SHALL BE PERFORMED WITHIN A FOUR HOUR PERIOD. THIS REQUIREMENT DOES NOT APPLY TO THE FUEL BUILDING EXHAUST.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 1 hour and less than or equal to 2% by volume within 48 hours. 6

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-12 of Specification 3.3.3.9.

PROOF AND REVIEW

AUG 1984 RECEIVED NUCLEAR OPERATIONS LICENSING

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 170,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 7 days when radioactive materials are being added to the tank and the quantity of radioactivity contained in the tank is less than or equal to one-half of the above limit; otherwise, determine the quantity of radioactive material contained in the tank at least once per 24 hours. DURING ADDITION.

PROOF AND REVIEW

AUG 1984
RECEIVED
NUCLEAR
OPERATIONS
LICENSING

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability.

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

PROOF AND REVIEW

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4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

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6.1 RESPONSIBILITY

6.1.1 The Director of Nuclear Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor, or during his absence from the Control Room, a designated SRO, shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice President-Nuclear Production shall be reissued to ~~all station personnel~~ on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1. OR AS SPECIFIED IN THE FSAR.

UNIT STAFF

6.2.2.1 The unit organization shall be as shown in Figure 6.2-2 OR AS SPECIFIED IN THE FSAR and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

c. A radiation protection technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Team of at least five members shall be maintained onsite at all times*. The Fire Team shall not include the Shift Supervisor, the STA, nor the (2) other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

6.2.2.2 The unit staff working hours shall be as follows:

a. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., Senior Reactor Operators, Reactor Operators, radiation protection technicians, auxiliary operators, and key maintenance personnel.

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*The radiation protection technician and Fire Team composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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UNIT STAFF (Continued)

b. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed(THIS EXCLUDES THE STA WORKING HOURS):

1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

c. Any deviation from the above guidelines shall be authorized by the Director of Nuclear Operations or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Director of Nuclear Operations or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

PROOF AND REVIEW

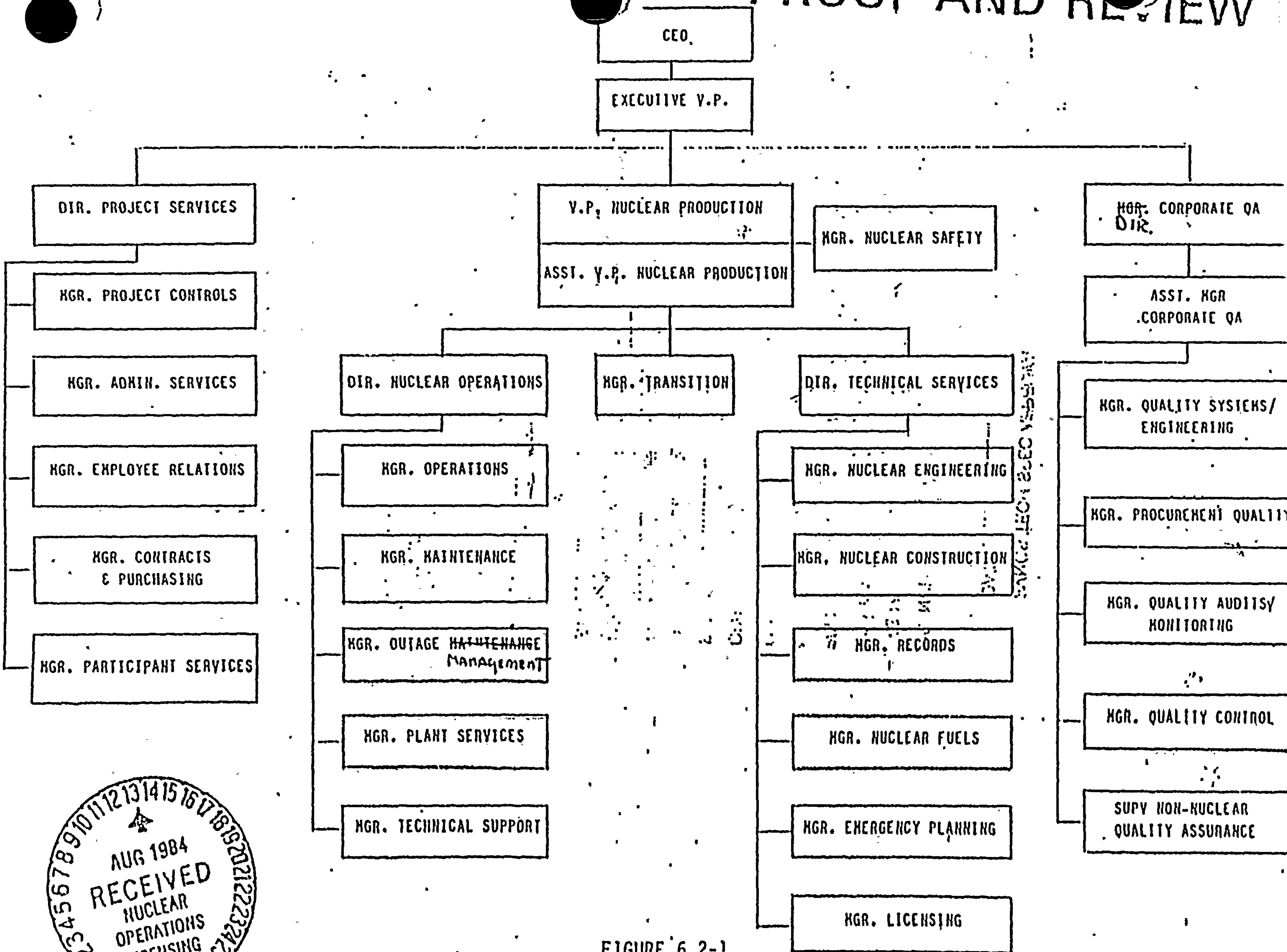

FIGURE 6.2-1
OFFSITE ORGANIZATION

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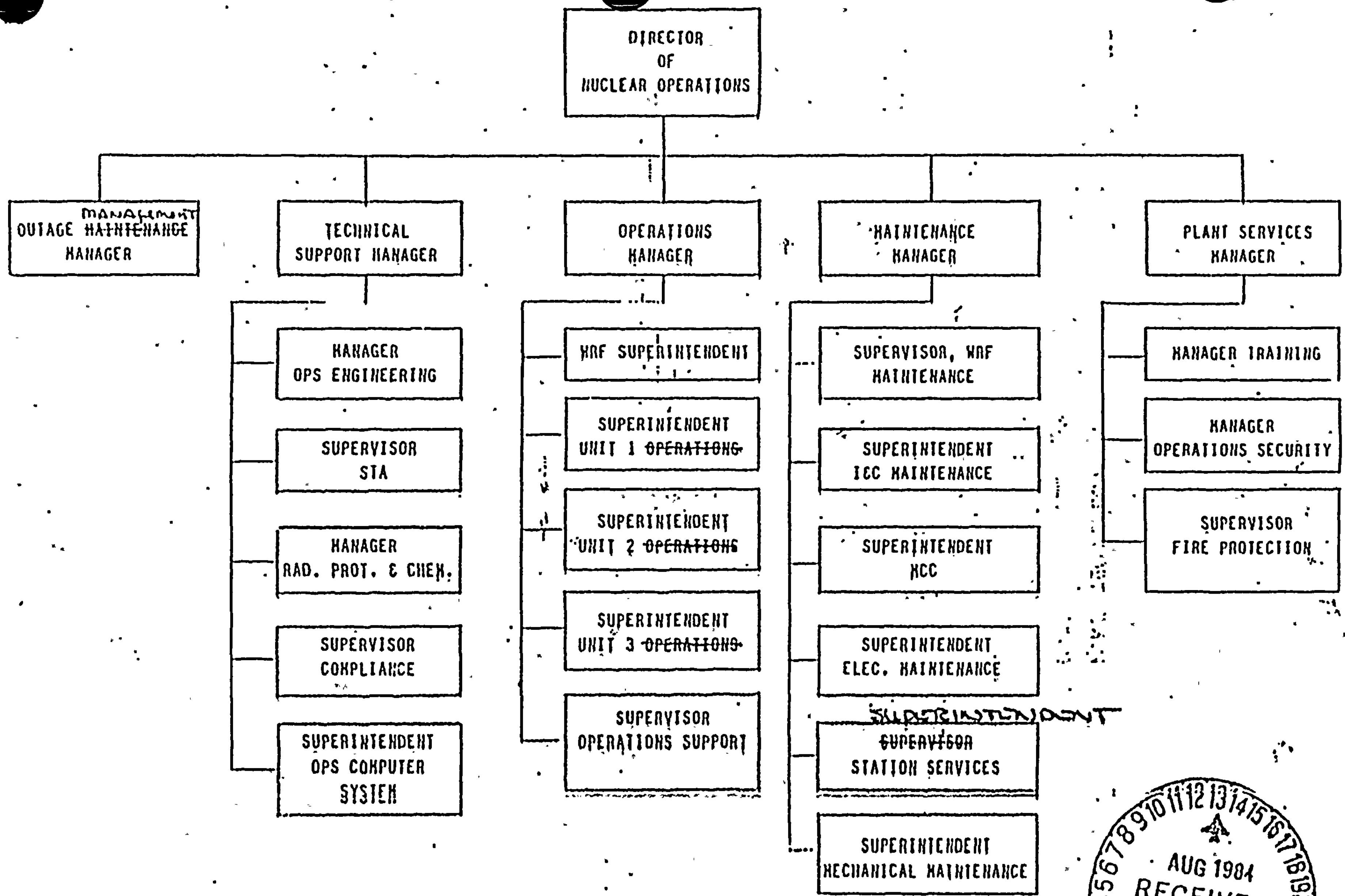

FIGURE 6.2-2

ONSITE UNIT ORGANIZATION

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Nuclear Operator I or II
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the Control Room command function.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 ~~The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.~~

Staff
See Attachment F

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Manager of Nuclear Safety, Director of Nuclear Operations, and the Supervisor, Nuclear Safety Group (NSG).

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Manager of Nuclear Safety. ~~and Supervisor of the NSG.~~

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1-1978, as endorsed by Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

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*Not responsible for sign-off function.

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6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Director of Nuclear Operations or his designee and shall meet or exceed the requirements and recommendations of Section 5.5 of ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The Plant Review Board shall function to advise the Director of Nuclear Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of the following personnel:

Member:	Technical Support Manager
Member:	Operations Manager
Member:	Maintenance Manager
Member:	Plant Services Manager
Member:	Engineering Manager
Member:	U1, U-2, U-3 ~~Operations~~ Superintendents
Member:	STA Supervisor
Member:	~~Training Manager~~ I and C Superintendent
Member:	Radiation Protection and Chemistry Manager
Member:	Quality Systems/Engineering Manager

The Director of Nuclear Operations shall designate the Chairman and Vice-Chairman in writing.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairman, or his designated alternate.

QUORUM minimum

6.5.1.5 The minimum quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, Vice-Chairman, or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The PRB shall be responsible for:

g. a. Review of ~~(1)~~ all Administrative control procedures ~~required by Specification 6.8 and changes thereto, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Director of Nuclear Operations, or designated alternate that affect Nuclear safety.~~ and changes

~~b. Review of all proposed tests and experiments that affect nuclear safety.~~

h. c. Review of all proposed changes to Appendix "A" Technical Specifications.

~~d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.~~

a. e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the ~~Director of Nuclear Operations, or designated alternate and to the~~ Nuclear Safety Group (NSG).

b. f. Review of ~~all REPORTABLE~~ EVENTS requiring 24-hour written notification to the Commission.

c. g. Review of unit operations to detect potential nuclear safety hazards.

d. h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Director of Nuclear Operations. ~~or designated alternate or the NSG.~~

~~i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Director of Nuclear Operations with copies to the NSG.~~

~~j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Director of Nuclear Operations with copies to the NSG.~~

f. k. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

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RESPONSIBILITIES (Continued)

e. ~~l.~~ Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

~~m. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Director of Nuclear Operations and to the NSG.~~

~~n. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and radwaste treatment systems.~~

AUTHORITY

6.5.1.7 The PRB shall:

~~a. Recommend in writing to the Director of Nuclear Operations, or his designated alternate approval or disapproval of items considered under Specification 6.5.1.6a. through d. above.~~

a. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. ~~through e.~~ above constitutes an unreviewed safety question.

b. Provide written notification within 24 hours to the Vice President-Nuclear Production and ~~the Supervisor of the~~ NSG of disagreement between the PRB and the Director of Nuclear Operations; however, the Director of Nuclear Operations ~~, or his designated alternate~~ shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1. above.

DNO

RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to ~~the Vice President Nuclear Production and the Supervisor of~~ the NSG.

~~COMMITTEES~~

~~6.5.1.9 Except for the review of programs and administrative control procedures, the PRB may establish standing committees to conduct reviews and make recommendations regarding the responsibilities identified in Specifications 6.5.1.6a., b., d., i., j., and 6.5.1.7a. and b. The following conditions must be met for such a committee:~~

~~a. The committee chairman shall be a PRB member designated in Specification 6.5.1.2;~~

~~b. The committee members shall be designated in writing by the PRB chairman;~~

Delete

~~RECORDS~~

~~6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Nuclear Safety Group.~~

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.2.1 The Director Nuclear Operations (DNO) shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Manager of Technical Support or the Manger of Engineering as previously designated by the Director of Nuclear Ops. Test results shall be approved by the Director of Nuclear Operations or the Manager of Technical Support.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the DNO; or by the Manager, Technical Support as previously designated by the DNO.

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the DNO to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the DNO, the Manager Technical Support, the Manager Operations, or the Manager Maintenance.

6.5.2.6 Review of the station security program, and implementing procedures, and submittal of recommended changes shall be approved by the DNO and transmitted to the Vice President-Nuclear Production and to the NSG.

PROOF AND REVIEW

ACTIVITIES (Continued)

6.5.2.7 Review of the station emergency plan, and implementing procedures, and submittal of recommended changes shall be approved.

6.5.2.8 The DNO shall assure the performance of a review by a qualified individual/organizaiton of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence.

6.5.2.9 The DNO shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

~~6.5.3 NUCLEAR SAFETY GROUP (NSG)~~

FUNCTION

6.5.3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. Nuclear power plant operations
b. Nuclear engineering
c. Chemistry and radiochemistry
d. Metallurgy
e. Instrumentation and control
f. Radiological safety
g. Mechanical and electrical engineering
h. Quality assurance practices

COMPOSITION

6.5.3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

New

~~COMMITTEES (Continued)~~

~~c. The responsibility, authority, and functions of the standing committees, including such matters as quorum requirements and documentation of committee activities shall be defined in administrative control procedures;~~

~~d. A report of the activities of each committee shall be made to the PRB at least once per calendar month; and~~

~~e. Any matters which cannot be resolved among committee members, or which the committee feels may involve a change to the technical specifications or an unreviewed safety question shall be forwarded to the PRB for review prior to implementation.~~

Delete

6.5.~~2~~3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.~~2~~3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices

COMPOSITION

6.5.~~2~~3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 6 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Four years experience in one of the designated areas in Specification 6.5.2.1. One of these 4 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.~~2~~3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

REVIEW

6.5.~~2~~3.4 The NSG shall be responsible program and its implementation for the review of:

a. The safety evaluations for (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question; This review will be done by the audit of selected safety evaluations.

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REVIEW (Continued)

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS requiring 24 hour notification;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the NSG or the Vice President-Nuclear Production.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

prepared monthly for the Manager, Nuclear Safety, who will

AUDITS (Continued)

~~i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.~~

~~j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.~~

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

AUTHORITY

6.5.~~2~~3.6 The NSG shall report to and advise the Manager of Nuclear Safety on those areas of responsibility specified in Specifications 6.5.~~2~~3.4 and 6.5.~~2~~3.5.

RECORDS

6.5.~~2~~3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be distributed ~~monthly~~ to the Vice President-Nuclear Production, Director of Nuclear Operations, ~~Manager of Nuclear Safety,~~ and to the management positions responsible for the areas audited.

6.6 REPORTABLE EVENT-ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS: Requiring 24 hour notification

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted to the ~~Supervisor of the NSG~~ Manager of Nuclear Safety and the Vice President-Nuclear ~~Production~~

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production, ~~the Supervisor of the NSG,~~ and Director of Nuclear Operations and Manager Nuclear Safety shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the Supervisor of the NSG and the Vice President-Nuclear Production within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC Addressable Constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB.

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5.1 and approved prior to implementation. Programs and administrative control procedures shall be approved by the Director of Nuclear Operations, or designated alternate. Implementing procedures shall be approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

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PROCEDURES AND PROGRAMS (Continued)

6.8.3 Temporary changes to procedures of Specification 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant ~~management~~ supervisory staff, at least one of whom ~~holds a Senior Reactor Operator's License~~ i.e. the Shift Supervisor or Assistant Shift Supervisor on the unit affected.

c. The change is documented, reviewed in accordance with Specification 6.5.1 and approved by the Director of Nuclear Operations or cognizant department head, as designated by the Director of Nuclear Operations, within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the NSG at least once per 24 months:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the high pressure safety injection system, the shutdown cooling portion of the low pressure safety injection system, the post-accident sampling subsystem of the reactor coolant sampling system, the containment spray system, ~~the post-accident sample return piping of the radioactive waste gas system~~, the post-accident sampling return piping of the liquid radwaste system, and the post-accident containment atmosphere sampling piping of the hydrogen monitoring subsystem. The program shall include the following:

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

Will be written and issued prior to system being declared operational

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

PROCEDURES AND PROGRAMS (Continued)

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(1) Training of personnel, and

(2) Procedures for monitoring.

e. Post-accident Sampling

Will be written and issued prior to system being declared operational

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencment of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,** e.g., reactor operations

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

ANNUAL REPORTS (Continued)

and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps**

*A single submittal may be made for a multiple unit station.

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive ~~liquid and~~ gaseous effluents released from the unit or station during the previous calendar year. ~~This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-3) during the report period.~~ All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. ~~The meteorological conditions concurrent with the time of release of radioactive materials in gaseous~~

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

~~effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. [For ORs, approximate and conservative approximate methods are acceptable.]~~ The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from ~~liquid and~~ gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous ~~and liquid~~ radioactive material released to the environs. X

e. Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.

f. Records of reactor tests and experiments.

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RECORD RETENTION (Continued)

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of PRB meetings ~~and of NSC activities~~.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

m. Records of audits performed under the requirements of Specifications 6.5.2.8 and 6.8.4.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

*Radiation Protection personnel or personnel escorted by Radiation Protection personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following plant radiation protection procedures for entry into high radiation areas.

HIGH RADIATION AREA (Continued)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Radiation Protection Supervisor or his designated alternate in the REP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area ~~and the maximum allowable stay time for individuals in that area~~. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems*, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information;

*Measurement made at 18 inches from source of radioactivity.

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PROCESS CONTROL PROGRAM (Continued)

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in ~~liquid and~~ gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in ~~liquid and~~ gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PRB.

b. Shall become effective upon review and acceptance by the PRB.

Enclosed are the branch's requested changes to the Palo Verde Unit 1, Proof and Review, Technical Specifications that were submitted to the Applicant for their review.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
A. Fuel Pool Area RU-31	1	**	≤5mR/hr	10^{-1} to 10^{4}mR/hr	22 & 24
B. New Fuel Area RU-19	1	*	≤5mR/hr	10^{-1} to 10^{4}mR/hr	22
C. Containment RU-148 & RU-149	2	1,2,3,4	≤~~2~~ 10 R/hr	1R/hr to 10^{7}R/hr	27
D. Containment Power Access Purge Exhaust RU-37 & RU-38	1	#	≤2.5mR/hr	10^{-1} to 10^{4}mR/hr	25
E. Main Steam					
1) RU-139 A&B	1	1,2,3,4	≤~~10~~ 3 mR/hr	10^{-3} to 10^{4}R/hr	27
2) RU-140 A&B	1	1,2,3,4	≤~~10~~ 3 mR/hr	10^{-3} to 10^{4}R/hr	27
2. Process Monitors					
A. Containment Building Atmosphere RU-1	2	1,2,3,4			23 & 27
1) Particulate			≤2.3×10^{-6}µCi/cc Cs-137	10^{-9} to 10^{-4}µCi/cc	
2) Gaseous			≤6.6×10^{-2}µCi/cc Xe-133	10^{-6} to 10^{-1}µCi/cc	
B. Noble Gas Monitors					
1) Control Room Ventilation Intake RU-29 & RU-30	1	All MODES	≤2×10^{-6}µCi/cc	10^{-7} to 10^{-1}µCi/cc	26

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TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
2) ~~Fuel Building Ventilation~~ Exhaust RU-145	1	**	Per ODCM	10^{-7} to 10^{-1} μCi/cc	27 & 24
3) Condenser Vacuum Pump/ Gland Seal Exhaust RU-141	1	1, 2, 3, 4	Per ODCM	10^{-7} to 10^{-1} μCi/cc	27
4) Plant Vent Gaseous RU-143	1	All MODES	Per ODCM	10^{-7} to 10^{-1} μCi/cc	27
5) Waste Gas Decay Tank Discharge RU-12	1	##	Per ODCM	10^{-3} to 10^{2} μCi/cc	***

**With fuel in the storage pool or building.
~~**With irradiated fuel in the storage pool.~~
***ACTION in accordance with Table 3.3-23 ACTION 35.
#When purge is being used.
~~##During waste gas release.~~

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TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
A. Fuel Pool Area RU-31	S	R	M	**
B. New Fuel Area RU-19	S	R	M	*
C. Containment Power Access Purge Exhaust RU-37 & RU-38	P#	R	P,W##	##
D. Containment RU-148 & RU-149	S	R	M	1,2,3,4
E. Main Steam RU-139 A&B RU-140 A&B	S	R	M	1,2,3,4
2. Process Monitors				
A. Containment Building Atmosphere RU-1				
1) Particulate	S	R	M	1,2,3,4
2) Gaseous	S	R	M	1,2,3,4
B. Noble Gas Monitors				
1) Control Room Ventilation Intake RU-29 & RU-30	S	R	M	All MODES
~~2) Fuel Building Ventilation Exhaust RU-145~~	~~S~~	~~R~~	~~M~~	~~**~~
~~3) Condenser Vacuum Pump/Gland Seal Exhaust RU-141~~	~~S~~	~~R~~	~~M~~	~~1,2,3,4~~

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4) Plant Vent RU-143	S	R	M	ALL MODES
5) Waste Gas Tanks Discharge RU-12	P	R	P	###

*With fuel in the storage pool or building.

~~**With irradiated fuel in the storage pool.~~

#If purge is in service for greater than 12 hours, perform once per 12-hour period.

##When purge system is in operation.

###During waste-gas release.

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor #RU-141 and #RU-142	1	1, 2, 3, 4	37
b. Iodine Sampler	1	1, 2, 3, 4	40
c. Particulate Sampler	1	1, 2, 3, 4	40
d. Flow Rate Monitor	1	1, 2, 3, 4	36
e. Sampler Flow Rate Measuring Device	1	1, 2, 3, 4	36
4. PLANT VENT SYSTEM			
a. Noble Gas Activity Monitor #RU-143 and #RU-144	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. FUEL BUILDING VENTILATION SYSTEM			
a. Noble Gas Activity Monitor #RU-145 and & RU-146	1	*	37, 41
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

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TABLE 4.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS RADWASTE SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (RU-12)	P	P	R(3)	Q(1)	*
b. Flow Rate Monitor	P	N.A.	R	Q	*
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (sequential)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (sequential)	D	N.A.	Q(5)	M	**

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this GASEOUS RADWASTE SYSTEM may continue for up to 14 days provided grab samples are collected at least once per 8 hours and are analyzed within the following 4 hours for the "on service" gas decay tank. With both channels inoperable, be in at least HOT STANDBY within 6 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2. within one hour after the channel has been declared inoperable ⊙

ACTION 41 - ~~With the number of channels operable less than requi~~ With the high range noble gas monitor ~~[illegible]~~ inoperable, comply with the ACTION requirement of Specification 3.9.12 (b) ⊙

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
3. CONDENSER EVACUATION SYSTEM (RU-141 & RU-142)					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	1, 2, 3, 4
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Flow Rate Monitor	D	N.A.	R	Q	1, 2, 3, 4
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	1, 2, 3, 4
4. PLANT VENT SYSTEM (RU-143 and RU-144)					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
5. FUEL BUILDING VENTILATION SYSTEM (RU-145 & RU-146)					
a. Noble Gas Actvity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R	Q	*