Attachment A

1. Remove pages - I, IV, IX, XI, XV, XVI, XVII, 1-4, 1-6, 1-7, B 3/4 3-3, 5-1, 5-2, 5-3, 6-10, 6-11, 6-12, 6-13, 6-14, 6-15, 6-16, 6-17, 6-18, 6-19, 6-20

2. Insert pages - I, Ia, IV, IVa, IX, IXa, XI, XIVa, XV, XVa, XVI, XVII, XVIIa, 1-4, 1-6, 1-7, 1-8, 1-9, 1-10, 3/4 3-53, 3/4 3-54, 3/4 3-55, 3/4 3-56, 3/4 3-57, 3/4 3-57a, 3/4 3-58, 3/4 3-59, 3/4 3-60, 3/4 3-61, 3/4 3-62, 3/4 3-63, 3/4 3-64, 3/4 3-65, 3/4 3-66, 3/4 3-67, 3/4 3-68, 3/4 11-1, 3/4 11-2, 3/4 11-3, 3/4 11-4, 3/4 11-5, 3/4 11-5a, 3/4 11-6, 3/4 11-7, 3/4 11-8, 3/4 11-9, 3/4 11-10, 3/4 11-11, 3/4 11-12, 3/4 11-13, 3/4 11-14, 3/4 11-15, 3/4 11-16, 3/4 11-17, 3/4 11-18, 3/4 11-19, 3/4 11-20, 3/4 11-21, 3/4 11-22, 3/4 11-23, 3/4 12-1, 3/4 12-2, 3/4 12-3, 3/4 12-4, 3/4 12-5, 3/4 12-6, 3/4 12-7, 3/4 12-8, 3/4 12-9, 3/4 12-10, 3/4 12-11, 3/4 12-12, B 3/4 3-3, B 3/4 3-4, B 3/4 11-1, B 3/4 11-2, B 3/4 11-3, B 3/4 11-4, B 3/4 11-5, B 3/4 11-6, B 3/4 12-1, 5-1, 5-1a, 5-1b, 5-1c, 5-1d, 5-1e, 5-2, 5-3, 6-10, 6-11, 6-12, 6-13, 6-14, 6-15, 6-16, 6-17, 6-18, 6-19, 6-20, 6-21, 6-22, 6-23, 6-24, 6-25, 6-26, 6-27, 6-28, 6-29, 6-30, 6-31

Attachment B

Proposed Technical Specification Change No. 36 Revision 3 involves incorporating the Radiological Effluent Technical Specification (RETS) into Appendix A. The Offsite Dose Calculation Manual and Appendix B are to be part of this Technical Specification Change Request and will be submitted at a later date.

The RETS provide additional requirements in the following areas:

1. Liquid and Gaseous Process Radiation Monitors are required to be operable during releases through their respective flow paths or alternate sampling frequencies/routines are established. Calibration requirements for outside radwaste storage tanks are set, attendant radiation and effluent monitoring functions/devices are required to be checked and calibrated. (Technical Specification 3.3.3.9 and 3.3.3.10).

2. Oxygen monitoring requirements are established for the waste gas system (Technical Specification 3.3.3.10 and 3.11.2.6)

3. Analysis sensitivities and frequencies for sampling are specified. Liquid and gaseous discharge concentrations for all effluent pathways must comply with 10 CFR 20 limits for discharge to unrestricted areas. An Offsite Dose Calculation Manual (ODCM) must be utilized to calculate release limits (Technical Specification 3.11.1.1 and 3.11.2.1).

4. Processing, reporting, sampling and ALARA goals for the liquid dose commitments to the public are specified. (Technical Specification 3.11.1.2 and 3.11.1.3).

5. Limits on the activities of liquid waste, gaseous tanks and sampling frequencies are specified to protect the public in the event of a failure (Technical Specification 3.11.1.4 and 3.11.2.5).

6. Processing, reporting, sampling and ALARA goals for the gaseous dose commitment to the public are specified (Technical Specification 3.11.2.1, 3.11.2.2, 3.11.2.3 and 3.11.2.4).

7. A solid waste monitoring program with procedures that complies with 10 CFR 20 and 10 CFR 71 requirements is implemented (Technical Specification 3.11.3.1).

8. A dose commitment to the public is established that considers all facility sources as potential contribution to the offsite dose (Technical Specification 3.11.4.1).

9. A Radiological Environmental monitoring program is established with sampling/routines for all potential pathways to the public (Technical Specification 3.12.1 and 3.12.2).

These proposed specifications do not affect the operability of any safety related components or structures. Incorporation of these specifications will not permit a significant change in effluent types or increase the quantities discharged, nor do they authorize an increase in power level. Therefore, pursuant to 10 CFR 41.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be submitted with this amendment.

Incorporation of this amendment will serve to reduce the quantity and activity of liquid waste discharged to the environment because of the incorporation of the Auxiliary Feedwater Pump Bay Drain Collection System. This system collects clean water, separates any entrained oil, monitors for activity and effectively segregates it from the liquid waste handling system. Automatic shutoff on high activity is provided and, therefore, the probability of an accidental release is not increased and does not affect the analysis or consequences of an accidental release as detailed in Section 14.2.2 of the FSAR. Segregating this water from the liquid waste system will provide improved processing capability since the steady state influent of waste water to the waste subsystems will be reduced.

This amendment will provide increased surveillance of effluent monitoring instrumentation, releases, waste handling systems and off-site monitoring and therefore should result in a reduction of the total dose to the public from gas and liquid releases.

This amendment will not increase the probability of occurrence of any accidents previously evaluated in Section 14 of the FSAR or introduce the possibility of a new accident. Therefore, the margin of safety as defined in the basis of the existing Technical Specification is not reduced and an unreviewed safety question does not exist.

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
DOCKET NO. 50-334
LICENSE DPR-66

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
Docket No. 50-334

License Amendment Application
To Incorporate Requirements of
10CFR50, Appendix I into the

BEAVER VALLEY POWER STATION
TECHNICAL SPECIFICATIONS

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

DEFINITIONS

SECTION

PAGE

1.0 DEFINITIONS

INDEX

DEFINITIONS

SECTION

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

INDEX

BASES

SECTION

PAGE

INDEX

DESIGN FEATURES

SECTION

PAGE

INDEX

DESIGN FEATURES

SECTION

PAGE

INDEX

ADMINISTRATIVE CONTROLS

SECTION

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one (1) excore detector inoperable, the remaining three (3) detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Regulatory Guide 1.109, 1977.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

b. The testing of one (1) system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

SOURCE CHECK

1.27 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

PROCESS CONTROL PROGRAM

1.28 A PROCESS CONTROL PROGRAM (PCP) shall be the manual or set of operating parameters detailing the program of sampling, analysis, and evaluation by which SOLIDIFICATION of wet radioactive wastes is assured. Requirements of the PCP are provided in Specification 6.14.

SOLIDIFICATION

1.29 SOLIDIFICATION shall be the conversion of wet radioactive wastes into a f [illegible] meets shipping and burial ground requirements.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.30 An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints. Requirements of the ODCM are provided in Specification 6.15.

GASEOUS RADWASTE TREATMENT SYSTEM

1.31 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

PURGE-PURGING

1.33 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is required to purify the confinement.

VENTING

1.34 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating conditions, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

MAJOR CHANGES

1.35 MAJOR CHANGES to radioactive waste systems, as addressed in Paragraph 6.16.2, (liquid, gaseous and solid) shall include the following:

1) Major changes in process equipment, components, structures and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2) Major changes in the design of radwaste treatment systems (liquid, gaseous and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

MEMBER(S) OF THE PUBLIC

1.36 MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries and persons who traverse portions of the site as the consequence of a public highway, railway, or waterway located within the confines of the site boundary. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 350°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 350°F$
3. HOT STANDBY	< 0.99	0	$\geq 350°F$
4. HOT SHUTDOWN	< 0.99	0	$350°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.

** Reactor vessel head unbolted or removed and fuel in the vessel.

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
P	Completed prior to each release.
N.A.	Not applicable.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of the radiation monitoring channels shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: During releases through the flow path.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or correct the alarm/trip setpoint.

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.3-12 or conservatively reduce the alarm setpoint. Exert a best effort to return the channel to operable status within 30 days, and if unsuccessful, explain in the next Semi-Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid 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 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Gross Activity Monitors Providing Automatic Termination of Release		
a. Liquid Waste Effluents Monitor (RM-LW-104)	(1)	23
b. Liquid Waste Contaminated Drain Monitor (RM-LW-116)	(1)	23
c. Auxiliary Feed Pump Bay Drain Monitor (RM-DA-100)	(1)	24
2. Gross Activity Monitors Not Providing Termination of Release		
a. Component Cooling-Recirculation Spray Heat Exchangers River Water Monitor (RM-RW-100)	(1)	24
3. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line	(1)	25
(1) FR-LW-103/RM-LW-116 (2) FR-LW-104/RM-LW-104		
b. Cooling Tower Blowdown Line	(1)	25
(1) FT-CW-101 (2) FT-CW-101-1		

TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
4.	Tank Level Indicating Devices (For tanks outside plant building)		
	a. Primary Water Storage Tank (BR-TK-6A)	(1)	26
	b. Primary Water Storage Tank (BR-TK-6B)	(1)	26
	c. Steam Generator Drain Tank (LW-TK-7A)	(1)	26
	d. Steam Generator Drain Tank (LW-TK-7B)	(1)	26

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may be resumed provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with specification 4.11.1.1.3, and;

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that at least once per 8 hours grab samples are analyzed for gross radioactivity (beta or gamma) at a Lower Limit of Detection (LLD) of at least 10^{-7} μCi/ml.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Line (RM-LW-104)	D	P(5)	R(3)	Q(1)
b. Liquid Waste Contaminated Drain Line (RM-LW-116)	D	P(5)	R(3)	Q(1)
c. Auxiliary Feed Pump Bay Drain Monitor (RM-DA-100)	D	D	R(3)	Q(1)
2. Gross Beta or Gamma Radioactivity Monitors Providing Alarm but not providing Automatic Termination of Release				
a. Component Cooling-Recirculation Spray Heat Exchangers River Water Monitor (RM-RW-100)	D	M(5)	R(3)	Q(2)
3. Flow Rate Monitors				
a. Liquid Radwaste Effluent Lines	D(4)	NA	R	Q
(1) FR-LW-103/RM-LW-116				
(2) FR-LW-104/RM-LW-104				
b. Cooling Tower Blowdown Line (FT-CW-101, 101-1)	D(4)	NA	R	Q

TABLE 4.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING
INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. Tank Level Indicating Devices (For tanks outside plant buildings)				
a. Primary Water Storage Tank - (BR-TK-6A)	D*	NA	R	Q
b. Primary Water Storage Tank - (BR-TK-6B)	D*	NA	R	Q
c. Steam Generator Drain Tank - (LW-TK-7A)	D*	NA	R	Q
d. Steam Generator Drain Tank - (LW-TK-7B)	D*	NA	R	Q

* During liquid additions to the tank.

TABLE 4.3-12 (Continued)

TABLE NOTATION

(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 exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Downscale failure.
3. 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 exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Downscale failure.
3. Instrument controls are not set in operate mode.

(3) - The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months. This can normally be accomplished during refueling outages. (Existing plants may substitute previously established calibration procedures for this requirement).

(4) - CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once daily on any day on which continuous, periodic, or batch releases are made.

(5) - A source check may be performed utilizing the installed means or flashing the detector with a portable source to obtain an upscale increase in the existing count rate to verify channel response.

RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 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 3.11.2.1 are not exceeded. The alarm/trip setpoints of the radiation monitoring channels shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: During releases through the flow path.

ACTION:

a. With a radioactive gaseous process or effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of 3.11.2.1 are met, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel or correct the alarm/trip setpoint.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.3-13 or conservatively reduce the alarm setpoint. Exert a best effort to return the channel to operable status within 30 days, and if unsuccessful, explain in the next Semi-Annual Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10.1 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 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
1. Gaseous Waste/Process Vent System (RM-GW-108A & B)				
a. Noble Gas Activity Monitor	(1)	*	Radioactivity Rate Measurement	27, 30***
b. Iodine Sampler Cartridge	(1)	*		32
c. Particulate Activity Monitor	(1)	*		32
d. System Effluent Flow Rate Measuring Device (FR-GW-108)	(1)	*	System Flow Rate Measurement	28
e. Sampler Flow Rate Measuring Device	(1)	*	Sampler Flow Rate Measurement	28
2. Auxiliary Building Ventilation System (RM-VS-101A & B)				
a. Noble Gas Activity Monitor	(1)	*	Radioactivity Rate Measurement	29, 30***
b. Iodine Sampler Cartridge	(1)	*		32

* During Releases via this pathway.

*** During purging of Reactor Containment via this pathway.

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
c.	Particulate Activity Monitor	(1)	*		32
d.	System Effluent Flow Rate Measuring Device (FR-VS-101)	(1)	*	System Flow Rate Measurement	28
e.	Sampler Flow Rate Measuring Device	(1)	*	Sampler Flow Rate Measurement	28
3.	Reactor Building/Supplementary Leak Collection and Release System (RM-VS-107A & B)				
a.	Noble Gas Activity Monitor	(1)	*	Radioactivity Rate Measurement	29, 30***
b.	Iodine Sampler Cartridge	(1)	*		32
c.	Particulate Activity Monitor	(1)	*		32
d.	System Effluent Flow Rate Measuring Device (FR-VS-112)	(1)	*	System Flow Rate Measurement	28
e.	Sampler Flow Rate Measuring Device	(1)	*	Sampler FLow Rate Measurement	28

* During Releases via this pathway.

*** During purging of Reactor Containment via this pathway.

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	PARAMETER	ACTION
4. Waste Gas Decay Tanks Monitor				
a. Oxygen Monitor (O_2-AS-GW-110-1,2)	(2)	**	Oxygen	31
b. Radiation Monitor (RM-GW-101)	(1)	**	Gross Activity	35
c. Sampler Flow Rate Measuring Device	(1)	**	Sampler Flow Rate Measurement	28

** During waste gas decay tank filling operation.

TABLE 3.3-13 (Continued)

TABLE NOTATION

ACTION 27 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank may be released to the environment provided that prior to initiating the release:

1. At least two independent samples of the tank's content are analyzed, and
2. 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 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 30 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, immediately suspend PURGING of Reactor Containment via this pathway if both RM-VS-104A and B are not operable with the purge/exhaust system in service.

TABLE 3.3-13 (Continued)

TABLE NOTATION

ACTION 31	-	With the number of channels OPERABLE one less than required by the MINIMUM Channels OPERABLE requirement, operation of this system may continue provided grab samples are obtained every 4 hours and analyzed within the following 4 hours during additions to a tank.
ACTION 32	-	With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 or sampled and analyzed once every 8 hours.
ACTION 35	-	See Surveillance 4.11.2.5.1.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gaseous Waste/Process Vent System (RM-GW-108A & B)				
a. Noble Gas Activity Monitor	P	P(5)	R(3)	Q(1)
b. Iodine Sampler Cartridge	W(6)	N/A	N/A	N/A
c. Particulate Activity Monitor	W	N/A	N/A	N/A
d. System Effluent Flow Rate Measuring Device (FR-GW-108)	P	N/A	R	Q
e. Sampler Flow Rate Measuring Device	D*	N/A	R	Q
2. Auxiliary Building Ventilation System (RM-VS-101A & B)				
a. Noble Gas Activity Monitor	D	M(5), P(5)***	R(3)	Q(2)
b. Iodine Sampler Cartridge	W(6)	N/A	N/A	N/A
c. Particulate Activity Monitor	W	N/A	N/A	N/A
d. System Effluent Flow Rate Measurement Device (FR-VS-101)	D	N/A	R	Q

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
e. Sampler Flow Rate	D	N/A	R	Q
3. Reactor Building/Supplementary Leak Collection and Release System (RM-VS-107A & B)				
a. Noble Gas Activity Monitor	D	M(5), P(5)***	R(3)	Q(1)
b. Iodine Sampler Cartridge	W(6)	N/A	N/A	N/A
c. Particulate Activity Monitor	W	N/A	N/A	N/A
d. System Effluent Flow Rate Measuring Device (FR-VS-112)	D	N/A	R	Q
e. Sampler Flow Rate Measuring Device	D	N/A	R	Q
4. Waste Gas Decay Tanks Monitor				
a. Oxygen Monitor (O_2-AS-GW-110-1,2)	D	N/A	Q(4)	M
b. Radiation Monitor (RM-GW-101)	D**	M(5)	R(3)	Q
c. Sampler Flow Rate Measuring Device	D**	N/A	R	Q

TABLE 4.3-13 (Continued)

TABLE NOTATION

* During releases via this pathway.

** During Waste Gas Tank filling operations.

*** During purging of Reactor Containment via this pathway.

(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 exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Downscale failure.
3. 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 exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Downscale failure.
3. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards certified by the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration should be used, at intervals of at least once per eighteen months. This can normally be accomplished during refueling outages. (Existing plants may substitute previously established calibration procedures for this requirement.)

TABLE 4.3-13 (Continued)

TABLE NOTATION

(4) 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.

(5) A source check may be performed utilizing the installed means or flashing the detector with a portable source to obtain an upscale increase in the existing count rate to verify channel response.

(6) Compliance is demonstrated in Table 4.11-2.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released at anytime from the site (See Figure 5.1-2) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} μCi/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released from the site to unrestricted areas exceeding the above limits; immediately restore concentration within the above limits.

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

c. Prompt notification pursuant to specification 6.9.1.8 is required.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1*.

4.11.1.1.2 The results of radioactive analysis shall be used in accordance with the methods of the ODCM to assure that the concentration at the point of release are maintained within the limits of specification 3.11.1.1.

* Radioactive liquid discharges are normally via batch modes. Turbine Building Drains shall be monitored as specified in Section 4.11.1.1.3.

SURVEILLANCE REQUIREMENTS (Continued)

4.11.1.1.3 When the activity of the secondary coolant is greater than 10^{-5} μCi/ml gross, grab samples shall be taken for each sump discharge from the turbine building and chemical waste sumps. The sample shall be analyzed for gross activity at a sensitivity of at least 10^{-7} Ci/ml and recorded in plant records. Water volume discharged shall be estimated from the number of pump operations unless alternate flow or volume instrumentation is provided.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μ Ci/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emmiters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e,g]	Grab Sample[g]	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	Grab Sample[g]	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Grab Sample[g]	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Grab Sample[g]	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample 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):

$$LLD = \frac{4.66\ s_b}{(E)\ (V)\ (2.22)\ (Y)\ \exp(-\lambda\Delta T)}$$

where

LLD is the lower limit of detection as defined above (as pCi 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 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;

ΔT is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent 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 shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-in milk samples). Typical values of E, V, Y and ΔT should be used in the calculations.

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. (Continued)

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

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

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

e. A continuous release exists when the discharge of liquid wastes is from a nondiscrete volume; e.g., from a volume of a system having an input flow during the continuous release. This is applicable to the Turbine Building drains and the AFW Pump Bay Drain System and chemical waste sump, when the secondary coolant gross radioactivity (beta and gamma) is greater than 10^{-5} μCi/ml.

f. The principal gamma emitters for which the LLD specification will apply are exclusively 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 detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should be reported as "less than" the nuclide's LLD, and should not be reported as being present at the LLD level for that nuclide. The "less than" values should not be used in the required dose calculations. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the semi-annual Radioactive Effluent Release Report.

TABLE 4.11-1 (Continued)

TABLE NOTATION

g. Whenever there is primary to secondary leakage, sampling is done for turbine building drain effluents by means of grab samples taken every 4 hours during the period of discharge and analyzed for gross radioactivity (beta and gamma) at a sensitivity of at least 10^{-7} μCi/ml and recorded in the plant records, along with the flow rate. Primary to secondary leakage is considered to be occurring whenever measurements indicate that secondary coolant gross radioactivity (beta and gamma) is greater than 10^{-5} μCi/ml. In addition, two (2) plant personnel shall check release calculations to verify that the limits of 3.11.1.1 and 3.11.1.2 are not exceeded.

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to MEMBER(S) OF THE PUBLIC FROM radioactive materials in liquid effluents released from the site (see Figure 5.1-2) shall be limited:

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

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem 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, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases, and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits. (This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act).*

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

---

* Applicable only if drinking water supply is taken from the receiving water body.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

LIQUID WASTE TREATMENT

LIMITING CONDITION OF OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be used to reduce the radioactive materials in each liquid waste batch prior to its discharge when the projected doses due to liquid effluent releases from the site (See Figure 5.1-2) when averaged over 31 days would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With liquid waste being discharged without treatment and exceeding the limits specified, in lieu of any other report required by specification 6.9.1, 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 operational status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

LIQUID WASTE TREATMENT (Continued)

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to $\leq$ 10 curies, excluding tritium and dissolved or entrained noble gases.

a. BR-TK-6A (Primary Water Storage Tank)

b. BR-TK-6B (Primary Water Storage Tank)

c. LW-TK-7A (Steam Generator Drain Tank)

d. LW-TK-7B (Steam Generator Drain Tank)

e. Miscellaneous temporary outside radioactive liquid storage tanks.

APPLICABILITY: At all times.

ACTICN:

a. With the quantity of radioactive material in any of the above listed tanks 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

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks 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.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in the unrestricted areas (see Figure 5.1-1) due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be ≤ 500 mrem/yr to the total body and ≤ 3000 mrem/yr to the skin, and

b. The dose rate limit, inhalation pathway only, for I-131, tritium and all radionuclides in particulate form (excluding C-14) with half-lives greater than 8 days shall be ≤ 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

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

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

c. Prompt notification pursuant to specification 6.9.1.8 is required.

SURVEILLANCE REQUIREMENTS

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

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.11.2.1.2 The dose rate, inhalation pathway only, for I-131, tritium and all radionuclides in particulate form (excluding C-14) with half-lives greater than 8 days. in gaseous effluents, shall be determined to be within the above limits in accordance with the methods and procedures the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

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[g]	1×10^{-4}
			H-3	1×10^{-6}
B. Containment Purge	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. Ventilation Systems 1. Process Vent 2. Containment Vent 3. Aux. Bldg. Vent	M[b,c,e] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[f]	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
Release from Radio-iodine and Particulates (Airborne) may be limited to the Inhalation Pathway only.	Continuous[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The Lower Limit of Detection (LLD) is defined in Table Notation (a) of Table 4.11-1 of Specification 4.11.1.1.

b. When reactor coolant system activity exceeds the limits stated in Specification 3.4.8, analyses shall be performed once every 24 hours during startup, shutdown and 25% load changes and 72 hours after achieving the maximum steady state power operation unless continuous monitoring is provided.

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

d. 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 and analyses shall also be performed at least once per 24 hours, during startup, shutdown and 25% load changes and 72 hours after achieving the maximum steady state power operation when RCS activity exceeds the limits in Specification 3.4.8 unless continuous monitoring is provided. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. 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.

f. The average 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 Specification 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification will apply are exclusively 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 measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the semi-annual effluent report.

RADIOACTIVE EFFLUENTS

DOSE, NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose in unrestricted areas (See Figure 5.1-1) due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta radiation.

b. During any calendar year, to ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Reprt which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2.1 Dose Calculations. Cumulative dose contributions shall be determined in accordance with the ODCM at least once every 31 days.

RADIOACTIVE EFFLUENTS

DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to MEMBER(S) OF THE PUBLIC from radioiodines and radioactive materials in particulate form (excluding C-14), and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents released from the site (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter to $\leq$7.5 mrem to any organ, and

b. During any calendar year to $\leq$15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioiodines, radioactive materials in particulate form, (excluding C-14), and radionuclides (other than noble gases) with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, which identifies the cause(s) for exceeding the limit and defines the corrective actions taken to reduce the releases and the proposed corrective actions to be taken to assure the subsequent releases will be within the above limits.

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3.1 Dose Calculations. Cumulative dose contributions shall be determined in accordance with the ODCM at least once every 31 days.

RADIOACTIVE EFFLUENTS

DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES (Continued)

Blank Page

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The Gaseous Radwaste Treatment 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 the site (see Figure 5.1-1), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of 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 the site (see Figure 5.1-1) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, 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 operational status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

SURVEILLANCE REQUIREMENTS (Continued)

BLANK PAGE

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.5 The quantity of radioactivity contained in each gas storage tank shall be limited to $\leq$ 52000 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

4.11.2.5.1 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank when the Waste Gas Decay Tank Monitor (RM-GW-101) is not operable and reactor coolant activity exceeds the limits of specification 3.4.8.

4.11.2.5.2 The Waste Gas Decay Tank Monitor (RM-GW-101) operability shall be determined in accordance with Table 4.3-13 unless sampling pursuant to 4.11.2.5.1 is being conducted.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of oxygen in the waste gas holdup system shall be limited to $\leq 2\%$ 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 $>2\%$ by volume, immediately suspend all additions of waste gases to the gaseous waste decay tank and reduce the concentration of oxygen to $\leq 4\%$ within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the affected tank and reduce the concentration of oxygen to less than or equal to 2% by volume within twelve hours.

c. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6.1 The concentrations of 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 oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10 or monitoring in conjunction with its associated action statement.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3.1 The solid radwaste system shall be used, as applicable, to solidify and package radioactive wastes, and to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71. Methods utilized to meet these requirements shall be described in facility procedures and in the Process Control Program (PCP).

APPLICABILITY: At all times.

ACTION:

a. With the applicable requirements of 10 CFR Part 20 and 10 CFR Part 71 not satisfied, suspend affected shipments of solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3.1.1 Prior to shipment, solidification shall be verified in accordance with Station Operating Procedures.

4.11.3.1.2 Reports. The semi-annual Radioactive Effluent Release Report in Specification 6.9.1.12 shall include the following information for each type of solid waste shipped offsite during the report period:

a. container volume;

b. total curie quantity (determined by measurement or estimate);

c. principal radionuclides (determined by measurement or estimate);

d. type of waste (e.g., spent resin, compacted dry waste evaporator bottoms);

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

f. solidification agent (e.g., cement, urea formaldehyde).

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4.1 The dose or dose commitment to MEMBER(S) OF THE PUBLIC from all facility releases is limited to ≤ 25 mrem to the total body or any organ (except the thyroid, which is limited to ≤75 mrem) for a calendar year.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 defining the corrective action and limit the subsequent releases such that the dose or dose commitment to MEMBER(S) OF THE PUBLIC is limited to ≤ 25 mrem to the total body or any organ (except thyroid, which is limited to ≤ 75 mrem) for a calendar year. This special report shall discribe the steps to be taken or modifications necessary to prevent a recurrence. Otherwise, obtain a variance from the Commission to permit releases which exceed the 40 CFR 190 Standard.

b. The provisions of Specification 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Dose Calculations. Cummulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, and 3.11.2.3.b, and in accordance with the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Environmental Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if speciments are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunction of automatic sampling equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.)

b. With the level of radioactivity in an environmental sampling medium at one or more of the locations specified in Table 3.12-1 exceeding the limits of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of affected calendar quarter a Report pursuant to Specification 6.9.1.9 which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits of Table 3.12-2 to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Report.

   When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration(1)}}{\text{Limit Level (1)}} + \frac{\text{Concentration (2)}}{\text{Limit Level (2)}} + \ldots \geq 1.0$$

3/4.12.1 MONITORING PROGRAM (Continued)

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued):

c. With milk or fresh leafy vegetable samples unavailable from the required number of locations selected in accordance with Specification 3.12.2 and listed in the ODCM, obtain replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1 and the ODCM provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations, if available.

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

SURVEILLANCE REQUIREMENTS

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type and Frequency[(a)] of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	5 Locations 1. One sample from a control location 10-20 miles distant and in the least prevalent wind direction. 2. One sample from vicinity of community having the highest calculated annual average ground level D/Q.	Continuous operation of sampler with sample collection at least weekly.	Each radioiodine canister. Analyze for I-131; Particulate sampler. Analyze for gross beta weekly[(b)]; Perform gamma isotopic analysis on composite (by location) sample at least quarterly.
2. DIRECT RADIATION	40 Locations. $\geq$2 TLD or a pressurized ion chamber at each location.	Continuous measurement with collection at least quarterly.	Gamma dose, quarterly.

(a) Analysis frequency same as sampling frequency unless otherwise specified.

(b) Particulate samples are not counted for $\geq$24 hours after filter change. Perform gamma isotopic analysis on each sample when gross beta is >10 times the yearly mean of control samples.

** Sample locations are given on figures and table in Offsite Dose Calculation Manual (ODCM).

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type and Frequency[(a)] of Analysis
3. WATERBORNE			
a. Surface	2 Locations. 1. One sample up-stream. 2. One sample down-stream.	Composite* sample collected over a period not to exceed one month.	Gamma isotopic analysis of each composite sample; Tritium analysis of composite sample at least quarterly.
b. Drinking	2 Locations.	Composite* sample collected over a period not to exceed 2 weeks.	I-131 analysis of each composite sample; Gamma isotopic analysis of composite sample (by location) monthly; Tritium analysis of composite sample. quarterly.
c. Groundwater	N/A - No wells in lower elevations between plant and river.		
d. Sediment from Shoreline	1 Location.	Semi-Annually	Gamma isotopic analysis semiannually.

* Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.

** Sample locations are shown on figures and tables in the Offsite Dose Calculation Manual (ODCM).

(a) Analysis frequency same as sampling frequency unless otherwise specified.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Locations**	Sampling and Collection Frequency	Type and Frequency[(a)] of Analysis
4. INGESTION			
a. Milk	4 Locations.[(c)] 1. Three samples selected on basis of highest potential thyroid dose using milch census data. 2. One local large dairy.	At least bi-weekly when animals are on pasture; at least monthly at other times.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	2 Locations.	Semi-Annual. One sample of available species.	Gamma isotopic analysis on edible portions.
c. Food Products (Leafy Vegetables)	4 Locations. 1. Three locations within 5 miles. 2. One control location.	Annually at time of harvest.	Gamma isotopic analysis and I-131 analysis on edible portion.

** Sample locations are shown on figures and tables in Offsite Dose Calculation Manual.

(a) Analysis frequency same as sampling frequency unless otherwise specified.

(c) Other dairies may be included as control station or for historical continuity. These would not be modified on basis of milch animal census.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water pCi/l	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Broad Leaf Vegetables (pCi/Kg, wet)
H-3	2×10^{4}(a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2			3×10^2	

(a) For drinking water samples. This is a 40 CFR Part 141 value

TABLE 4.12-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/l)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	1×10^{-2}				
H-3	2000					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-95	30[c]					
Nb-95	15[c]					
I-131	1[b]	7×10^{-2}		1	60	
Cs-134	15	5×10^{-2}	130	15	60	150
Cs-137	18	6×10^{-2}	150	18	80	180
Ba-140	60[c]			60		
La-140	15[c]			15		

NOTE: 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 be identified and reported.

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample 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):

$$LLD = \frac{4.66}{(E)\ (V)\ (2.22)\ (Y)\ \exp(-\lambda \Delta T)}$$

where

LLD is the lower limit of detection as defined above (as pCi 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 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;

ΔT is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent 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 shall 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 calculations.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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

b. LLD for drinking water.

c. If parent and daughter are totaled, the most restrictive LLD should be applied.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. (For elevated releases as defined in Regulatory Guide 1.111, (Rev. 1) July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.)

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, in lieu of any other report, which identifies the new location(s).

b. With a land use census identifying a milch animal location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report, in lieu of any other report, which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days, if possible. The milk sampling program shall include samples from the three active milch animal locations, having the highest calculated dose or dose commitment. Any replaced location may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted.

c. The provisions of Specification 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS (Continued)

SURVEILLANCE REQUIREMENTS

4.12.2.1 The land use census shall be conducted at least once per 12 months between the dates of (June 1 and October 1) using that information which will provide the best results, such as by a door-to-door survey,* aerial survey, or by consulting local agriculture authorities.

* Confirmation by telephone is equivalent to door-to-door.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.2 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program.

APPLICABILITY: At all times.

ACTICN:

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 Report.

b. The provisions of Specification 3.0.3, 6.9.1.9 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3.1 The results of analyses performed as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Report.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

3/4.3.3.8 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT INSTRUMENT

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENT

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposure within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to an individual and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April, 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

This specification applies to the release of liquid effluents from Beaver Valley Power Station, Unit No. 1. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE TREATMENT

The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents. This specification applies to Beaver Valley Power Station, Unit No. 1

3/4.11.1.4 LIQUID HOLDUP TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix A, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at anytime at the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the exclusion area boundary to 500 mrem/year to the total body or to 3,000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to 1,500 mrem/year for the nearest cow to the plant.

This specification applies to the release of gaseous effluents from Beaver Valley Power Station, Unit No. 1. For units with shared radwaste treatment system, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE, NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric

BASES

Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July, 1977. The ODCM equations provided for determining the air doses at the exclusion area boundary are based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111. This specification applies to the release of gaseous effluents from Beaver Valley Power Station, Unit No. 1.

3/4.11.2.3 DOSE, RADIOIODINES, RADIOACTIVE MATERIAL IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I.

The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A. of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routing Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision I, October, 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July, 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form, and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which are examined in the development of these calculations are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man. This specification applies to radioactive material in particulate form and radionuclides other than noble gases released from Beaver Valley Power Station, Unit No. 1

BASES

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents. This specification applies to gaseous radwaste from Beaver Valley Power Station, Unit No. 1

3/4.11.2.5 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting total body exposure to an individual located at the nearest exclusion area boundary for two hours immediately following the onset of the release will not exceed 0.5 rem. The specified limit restricting the quantity of radioactivity contained in each gas storage tank was specified to ensure that the total body exposure resulting from the postulated release remained a suitable fraction of the reference value setforth in 10 CFR 100.11 (a)(1).

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Isolation of the affected tank for purposes of purging and/or discharge permits the flammable gas concentrations of the tank to be reduced below the lower explosive limit in a hydrogen rich system. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

BASES

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A of 10 CFR Part 50 and requires the system be used whenever solid radwastes require processing and packaging prior to being shipped offsite. The process parameters used in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The Specification requires the preparation and submittal of a Special Report, in lieu of any other report, whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 nuclear reactors, it is highly unlikely that the resultant dose to MEMBER(S) OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to MEMBER(S) OF THE PUBLIC for the calendar year to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to MEMBER(S) OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

BASES

3/4.12.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of MEMBER(S) OF THE PUBLIC resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring programs are made if required by the results of this census. The best survey information from the door-to-door survey, aerial survey or by consulting with local agriculture authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an Interlaboratory Comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

5.0 DESIGN FEATURES

5.1 SITE

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.1 The site boundary for gaseous effluents shall be as shown in Figure 5.1-1. Release paths are shown on Figure 5.1-5.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.2 The site boundary for liquid effluents shall be as shown in Figure 5.1-2. Release points are shown on Figure 5.1-6.

EXCLUSION AREA

5.1.3 The exclusion area shall be as shown in Figure 5.1-3.

LOW POPULATION ZONE

5.1.4 The low population zone shall be as shown in Figure 5.1-4.

FLOOD CONTROL

5.1.5 The flood control provisions (dikes, levees, etc.) shall be designed and maintained in accordance with the original design provisions contained in Section 2.3.2.2 of the FSAR.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 185 feet.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

5.2 CONTAINMENT (Continued)

CONFIGURATION (Continued)

e. Minimum thickness of foundation mat = 10 feet.

f. Nominal thickness of vertical portion of steel liner = 3/8 inch.

g. Nominal thickness of steel liner, dome portion = 1/2 inch.

h. Net free volume = 1.8×10^6 cubic feet.

SITE BOUNDARY FOR GASEOUS EFFLUENTS
FOR THE BEAVER VALLEY POWER STATION

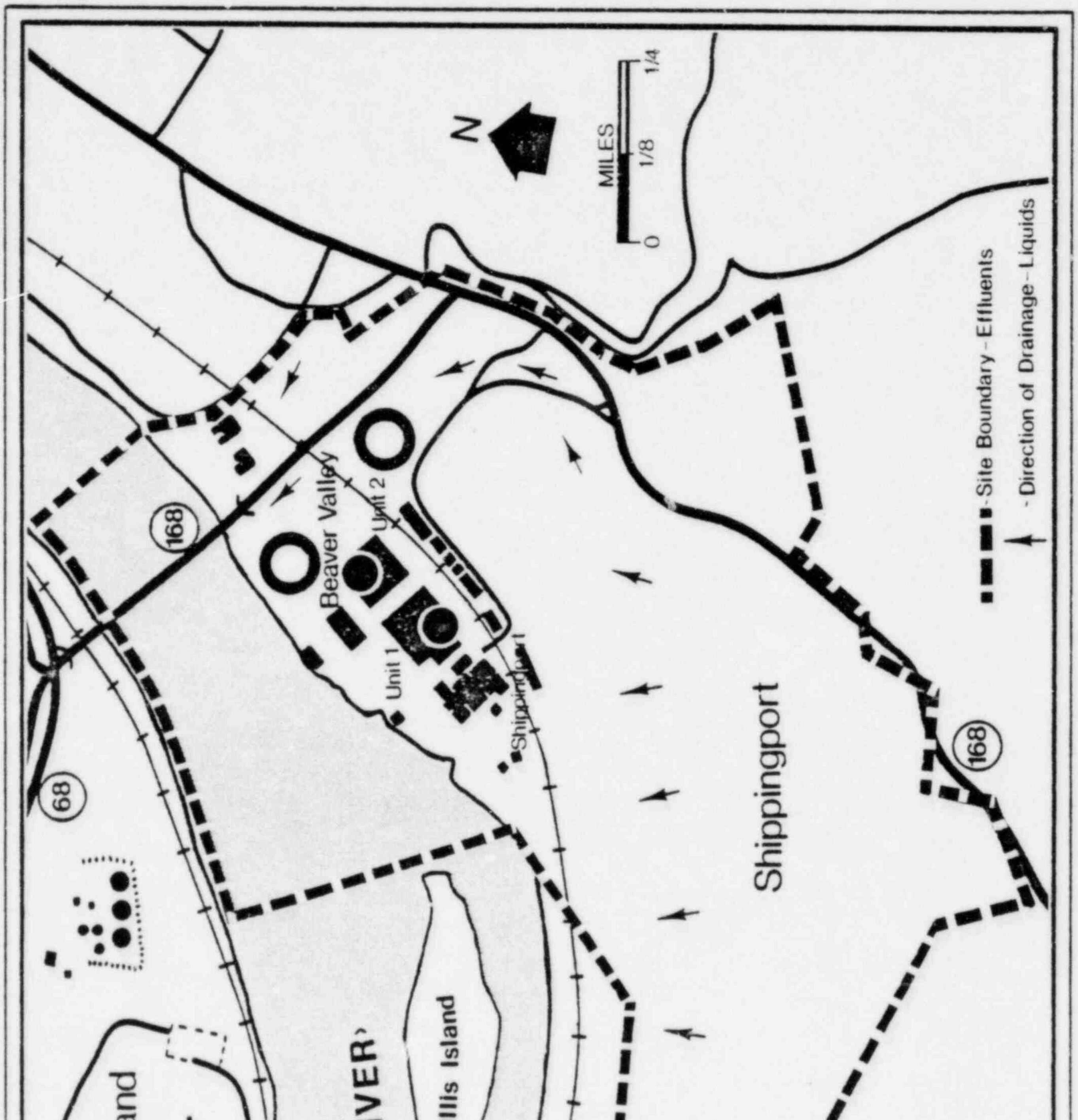


FIGURE 5.1-1

SITE BOUNDARY FOR LIQUID EFFLUENTS
FOR THE BEAVER VALLEY POWER STATION

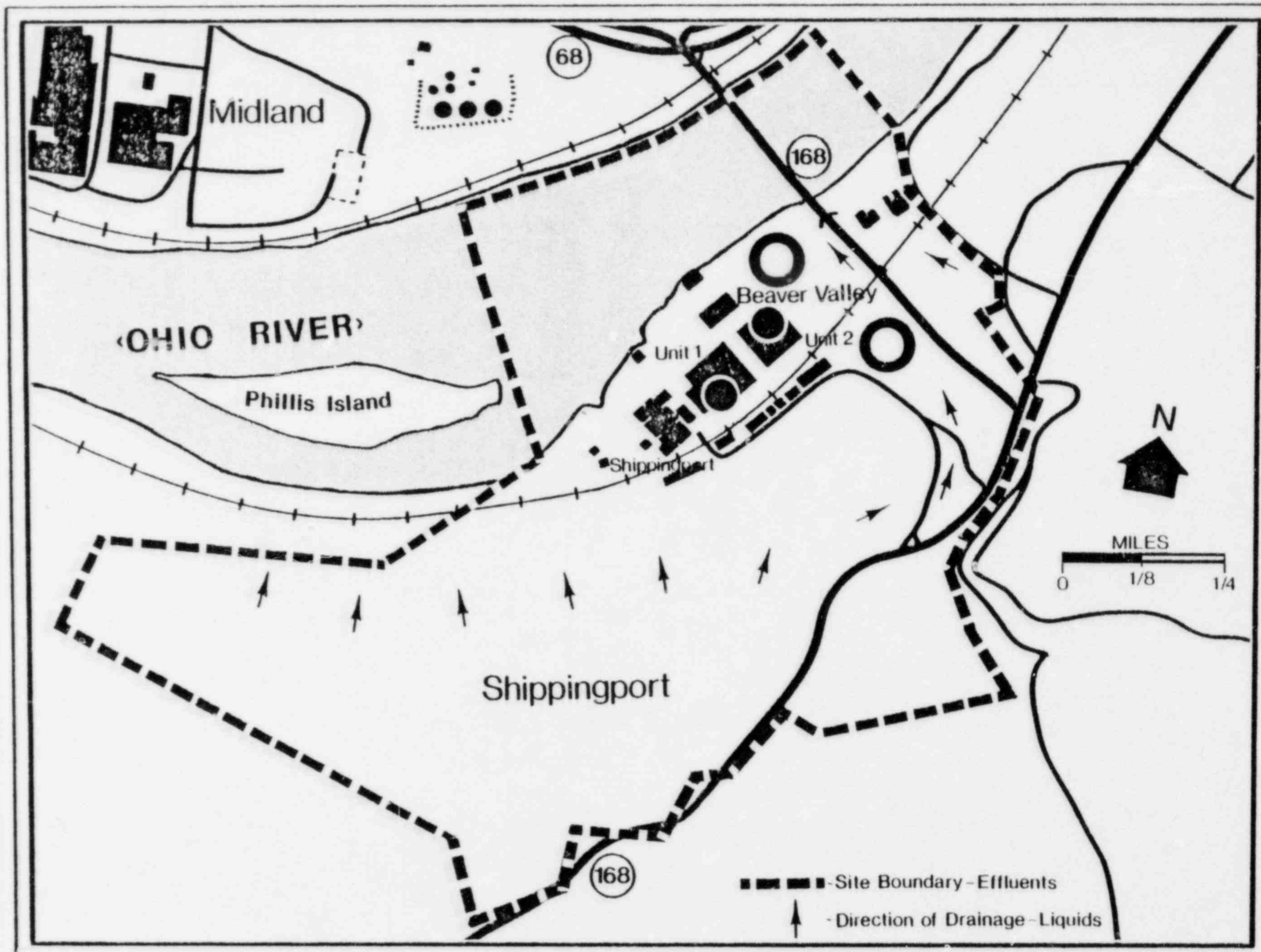
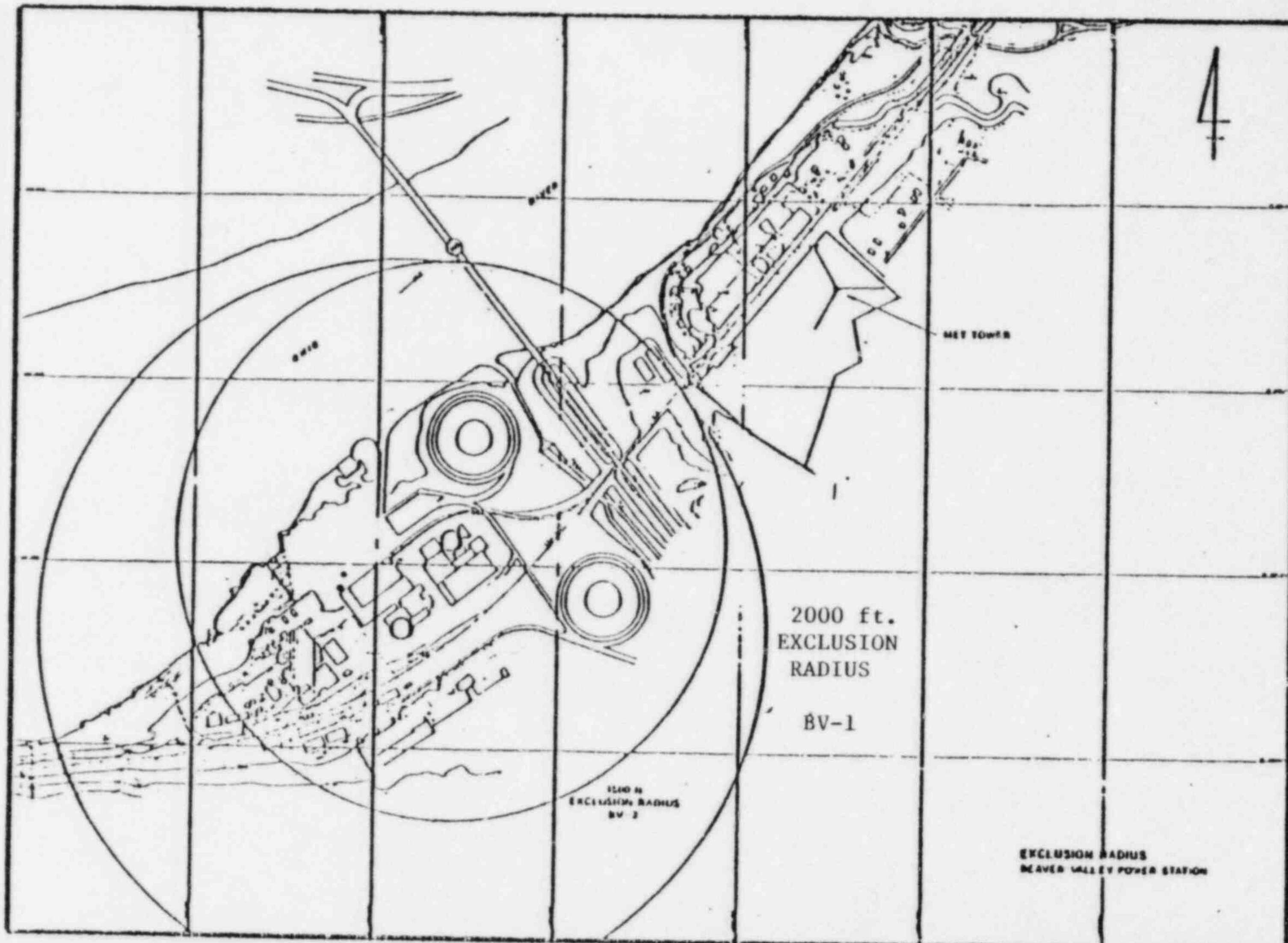
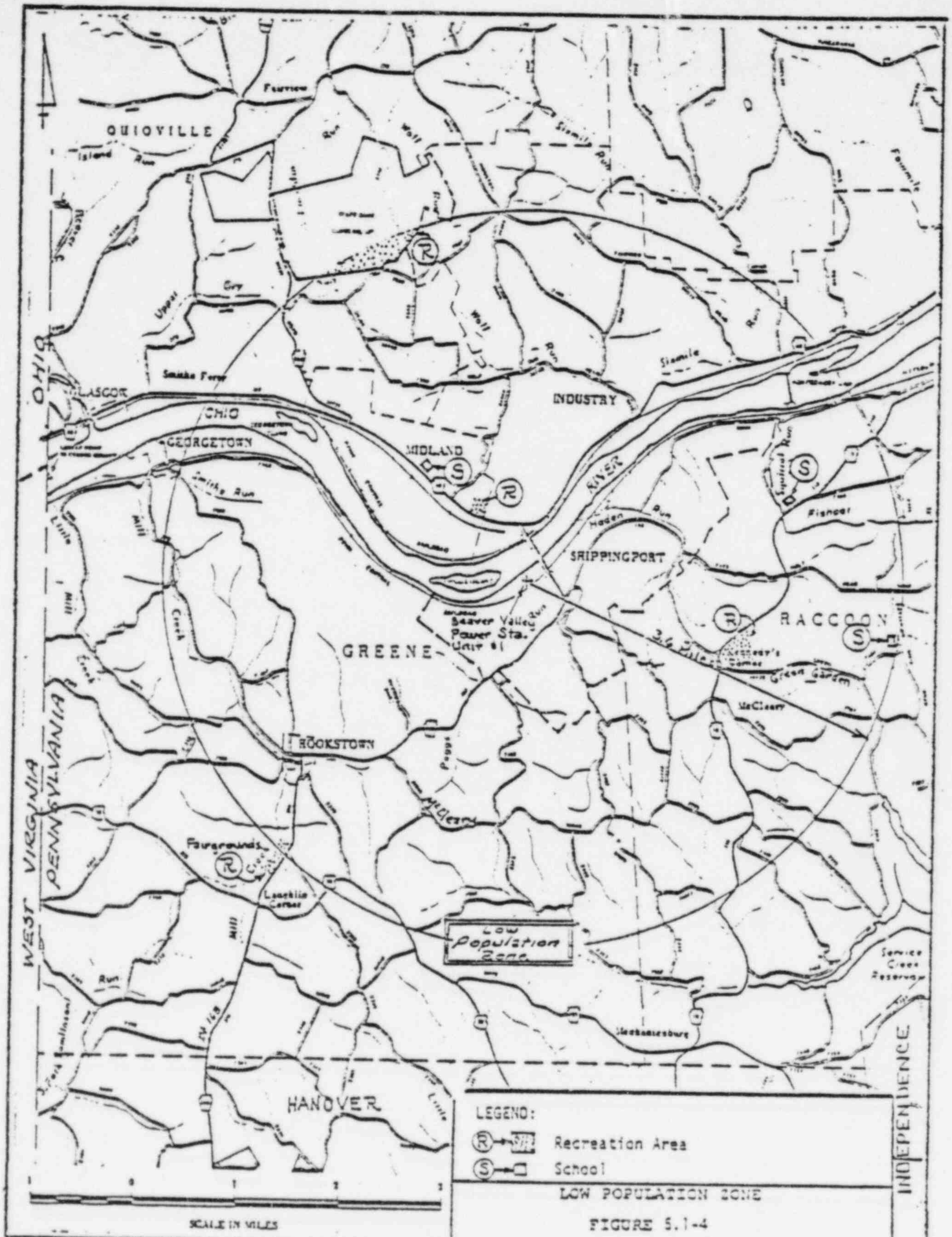


FIGURE 5.1-2



EXCLUSION AREA - BEAVER VALLEY POWER STATION

FIGURE 5.1-3



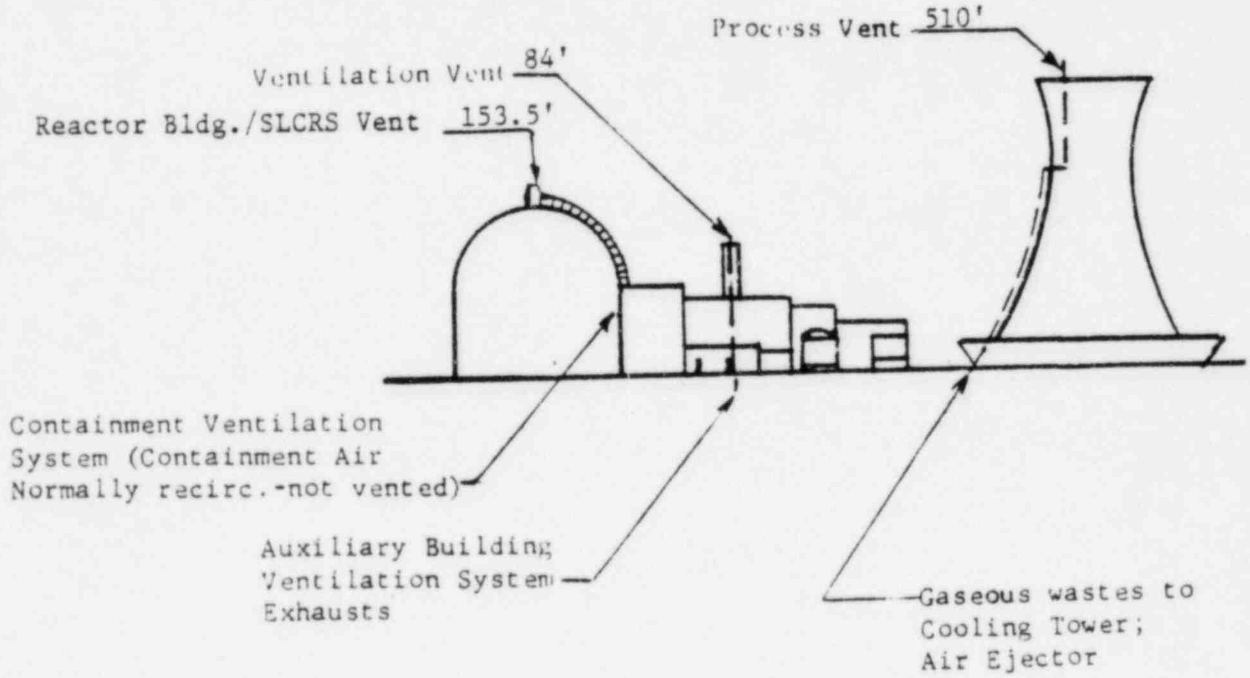
5-5

FIGURE 5.1-4

LOW POPULATION ZONE - BEAVER VALLEY POWER STATION

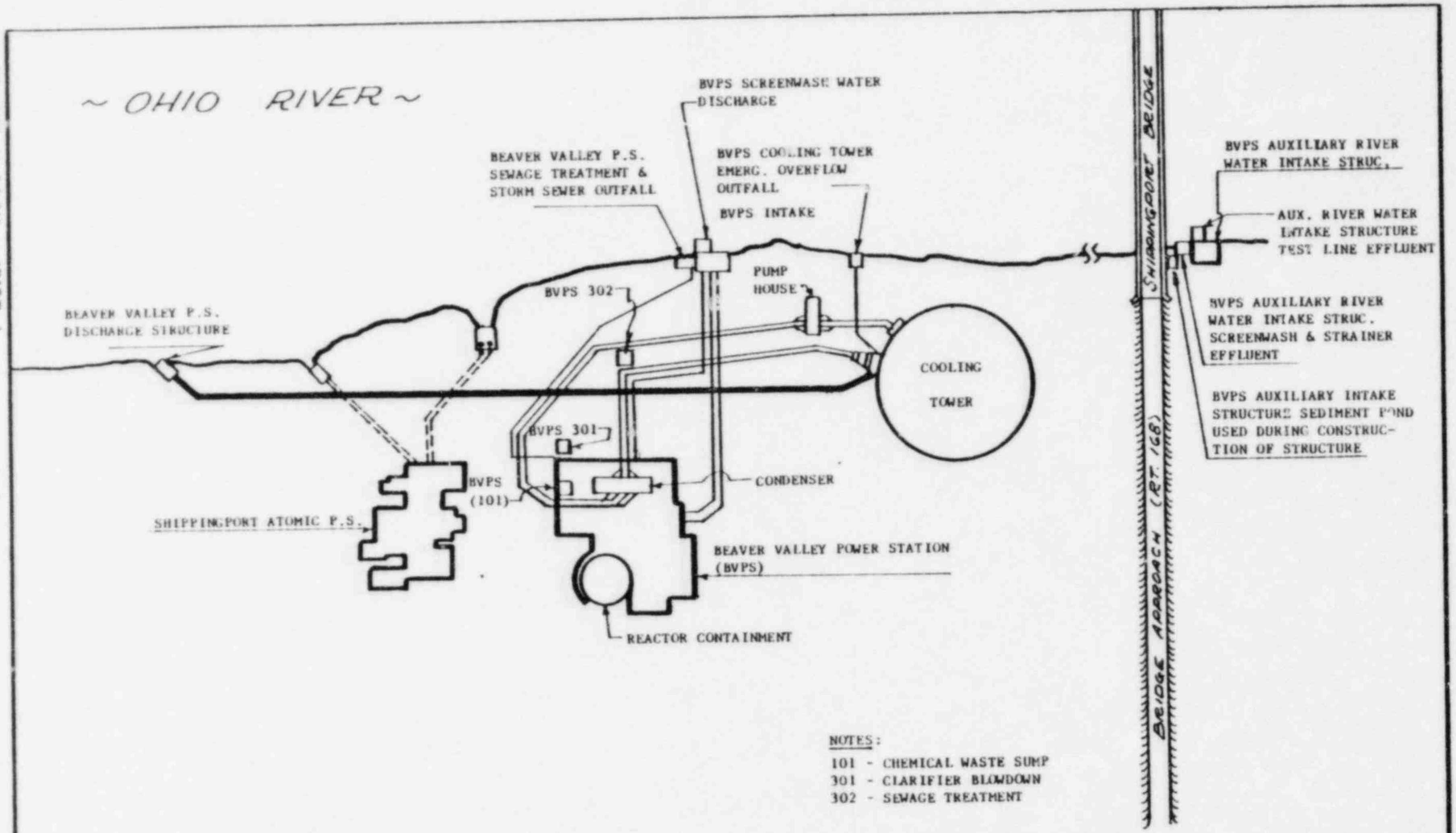
FIGURE 5.1-5

DISCHARGE POINTS - GASEOUS WASTES



GASEOUS RELEASE POINTS - BEAVER VALLEY POWER STATION

Figure 5.1-5



NOTES:

101 - CHEMICAL WASTE SUMP
301 - CLARIFIER BLOWDOWN
302 - SEWAGE TREATMENT

LIQUID RELEASE POINTS - BEAVER VALLEY POWER STATION

Figure 5.1-6

RIVER INTAKE AND DISCHARGE IN OHIO RIVER

SHIPPINGPORT & BEAVER VALLEY POWER STATIONS

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the ORC. These audits shall encompass:

a. The conformance of facility 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 facility staff at least oncer per 12 months.

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

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

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. Any other area of facility operation considered appropriate by the ORC or the Vice President, Nuclear.

h. The Facility Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified off-site licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

AUDITS (Continued)

AUTHORITY

6.5.2.9 The ORC shall report to the Manager of Nuclear Safety and Licensing and advise the Vice President, Nuclear Division on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of ORC activities shall be prepared, approved and distributed as indicated by the following:

a. Minutes of each ORC meeting shall be prepared, approved and forwarded to the Vice President, Nuclear Division within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President, Nuclear Division within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Vice President, Nuclear Division and to the management positions responsible for the areas audited within 30 days after completion of the audit.

d. The chairman of the ORC shall review all recommendations of the ORC and shall forward such recommendations to the Vice President, Nuclear Division.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the OSC and submitted to the ORC and the Manager of Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

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

a. The facility shall be placed in at least HOT STANDBY within one (1) hour.

b. The Safety Limit violation shall be reported to the Commission, the Manager of Nuclear Operations and to the ORC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the On-Site Safety Committee (OSC). 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.

d. The Safety Limit Violation Report shall be submitted to the Commission, the ORC and the Manager of Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

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, November, 1972.

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. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the OSC and approved by the Plant Superintendent, predesignated alternate or a predesignated Department Manager to whom the Plant Superintendent has assigned in writing the responsibility for review and approval of specific subjects considered by the committee, as applicable.

6.8.3 Temporary changes to procedures of 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 (2) members of the plant management staff, at least one (1) of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the OSC and approved by the Plant Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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 Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing will 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 requested 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 commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS [1]

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. A tabulation of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions[2] (e.g., reactor operations 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 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

---

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

2. This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, submitted no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety-system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety-system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a shutdown margin less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events, that, as a direct result of the event, require plant shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than that assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during plant life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe conditon.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

THIRTY-DAY WRITTEN REPORT

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the technical specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

NOTE: Routine surveillance testing, instrument calibration, or preventive maintenance which requires system configurations as described in items a. and b. above, need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above, designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

THIRTY-DAY WRITTEN REPORT (Continued)

e. (Continued)

1. A description of the event and equipment involved;

2. Cause(s) for the unplanned release;

3. Actions taken to prevent recurrence;

4. Consequences of the unplanned release.

f. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.5 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

g. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT [3]

6.9.1.10 Routine 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 and will include reporting any deviations not reported under 6.9.1.9 with respect to the Radiological Effluent Technical Specifications.

---

3. A single submittal may be made for a multiple unit site. The submittal should combine those sections that are common to both units.

ANNUAL RADIOLOGICAL ENVIRONMENTAL REPORT (Continued)

6.9.1.11 The annual radiological environmental reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and 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 the land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some 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.

TABLE 6.9-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility ______________________ Docket No. ______________

Location of Facility ______________________ Reporting Period ______________
(County, State)

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limits of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range[b]	Locations with Highest Annual Mean: Name Distance and Direction	Locations with Highest Annual Mean: Mean (f)[b] Range[b]	Control Locations Mean (f)[b] Range[b]	Number of REPORTABLE OCCURRENCES

[a] Nominal Lower Limit of Detection (LLD) as defined in table notation a. of Table 4.12-1 of Specification 4.12.1.1.

[b] Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses (f).

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program required by Specification 3.12.3.

SEMI-ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT [4]

6.9.1.12 Routine radioactive effluent release reports covering the operating 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.

6.9.1.13 The 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, Revision 1, June, 1974, "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," with data summarized on a quarterly basis following the format of Appendix B thereof.

In addition the radioactive effluent release report to be submitted 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 of wind speed, wind direction, atmospheric stability, and precipitation (if measured) on magnetic tape, 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 report shall also include an assessment of the radiation doses from radioactive effluents to MEMBER(S) OF THE PUBLIC due to their activities inside the site boundary (Figure 5.1-1 and 5.1-2) during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

---

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

The 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 real individual from reactor releases for the previous calendar year to show conformance with 40 CFR 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, Revision 1. The SKYSHINE Code (available from Radiation Shielding Information Center, ORNL) is acceptable for calculating the dose contribution from direct radiation due to N-16.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter as outlined in Regulatory Guide 1.21. In addition, the unrestricted area boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with the ODCM.

The radioactive effluent release reports shall also include any licensee initiated changes to the ODCM made during the 6 month period.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement (Regional Office) within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.

b. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

c. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

d. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

e. Seismic event analysis, Specification 4.3.3.3.2.

f. Sealed source leakage in excess of limits, Specification 4.7.9.1.3.

g. Fire Detection Instrumentation, Specification 3.3.3.6.

h. Fire Suppression Systems, Specifications 3.7.14.1, 3.7.14.2 and 3.7.14.3.

i. Miscellaneous reporting requirements specified in the Action Statements for Radiological Effluent Technical Specifications.

6.10 RECORD RETENTION

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

a. Records and logs of facility 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 OCCURRENCES submitted to the Commission.

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

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. 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 Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

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

c. Records of facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

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

h. Records of in-service 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 meetings of the OSC and the ORC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lines of all hydraulic and mechanical snubbers listed on Table 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

n. Records of analyses required by the Radiological Environmental Monitoring Program.

6.11 RADIATION PROTECTION PROGRAM

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 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 Radiological Work Permit* or Radiological Access Control Permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by a facility health physics supervisor in the Radiological Work Permit or Radiological Access Control Permit.

6.12.2 The requirements of 6.12.1, above, also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or a facility health physics supervisor.

---

* Health physics personnel, or personnel escorted by health physics personnel in accordance with approved emergency procedures, shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982, all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License No. DPR-66 dated October 24, 1980.

6.13.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.14 PROCESS CONTROL PROGRAM (PCP)

FUNCTION

6.14.1 The PCP shall be a manual containing the processing steps and a set of established process parameters detailing the program of sampling, analysis, and evaluation within which solidification of radioactive wastes is assured, consistent with Specification 3.11.3.1 and the surveillance requirements of these Technical Specifications.

6.14.2 Licensee initiated changes:

1. Shall become effective upon review and acceptance by the OSC.

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

FUNCTION

6.15.1 The ODCM shall describe the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

6.15.2 Licensee initiated changes:

1. Shall become effective upon review and acceptance by the OSC.

6.16 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

FUNCTION

6.16.1 The radioactive waste treatment systems (liquid, gaseous and solid are those systems described in the facility Final Safety Analysis Report or Hazards Summary Report, and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the LCO's set forth in Specifications 3.11.1.1, 3.11.1.2, 3.11.1.3, 3.11.1.4, 3.11.2.1, 3.11.2.2, 3.11.2.3, 3.11.2.4, 3.11.2.5, 3.11.2.6, 3.11.3.1, and 3.11.4.1.

6.16.2 Major changes as defined in Section 1 to the radioactive waste systems (liquid, gaseous and solid) shall be made by the following method:

A. Licensee initiated changes:

1. If a permanent facility change is made to a radioactive treatment system that could result in an increase in the volume or activity discharged, the Commission shall be informed by the inclusion of a suitable discussion of each change in the Annual 10 CFR 50.59 Report for the period in which the changes were made. The discussion of each change shall contain:

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

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

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

   d. An evaluation of the change will be submitted which shows the predicted increase of releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected increase in the maximum exposures to an individual in the unrestricted area from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted increase of releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period the changes were made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the OSC.

2. The change shall become effective upon review and acceptance by the OSC.

6.13.3 Background of what constitutes "major changes" to radioactive waste systems (liquid, gaseous, and solid).

A. Background

1. 10 CFR Part 50, Section 50.34a(a) requires that each application to construct a nuclear power reactor provide a description of the equipment installed to maintain control over radioactive material in gaseous and liquid effluents produced during normal reactor operations including operational occurrences.

2. 10 CFR Part 50, Section 50.34a(b)(2) requires that each application to construct a nuclear power reactor provide an estimate of the quantity of radionuclides expected to be released annually to unrestricted areas in liquid and gaseous effluents produced during normal reactor operation.

3. 10 CFR Part 50, Section 50.34a(3) requires that each application to construct a nuclear power reactor provide a description of the provisions for packaging, storage and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

4. 10 CFR Part 50, Section 50.34a(3)(c) requires that each application to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems and (2) a revised estimate of the information required in (b)(2) if the expected releases and exposures differ significantly from the estimate submitted in the application for a construction permit.

5. The Regulatory staff's Safety Evaluation Report and amendments thereto issued prior to the issuance of an operating license contains a description of the radioactive waste systems installed in the nuclear power reactor and a detailed evaluation (including estimated releases of radioactive materials in liquid and gaseous waste and quantities of solid waste produced from normal operation, estimated annual maximum exposures to an individual in the unrestricted area and estimated exposures to the general population) which shows the capability of these systems to meet the appropriate regulations.

6.17 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Manager of Nuclear Safety; and Licensing delegates the responsibility for the Radiological Environmental Monitoring Program to the Director, Environmental and Radiological Safety Programs (Figure 6.13.1) or his designated alternate.

The Director, Environmental and Radiological Safety Programs is responsible for administering the offsite Radiological Environmental Monitoring Program. He shall determine that the sampling program is being implemented as described to verify that the environment is adequately protected under existing procedures. He shall also have the responsibility for establishing, implementing, maintaining and approving offsite environmental program sampling, analyses and calibration procedures.