RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

REVIEW DOCUMENT

BRUNSWICK STEAM ELECTRIC PLANT

AUGUST 1982

NOTE: This document is for information only and is not a formal request for a Technical Specification change. It is intended to be used as a bases for reviews and discussions that will lead to the inclusion of Radiological Effluent Technical Specifications (RETS) into the Brunswick Steam Electric Plant's Technical Specifications. This document consists of revised excerpts from those sections of the Tech Specs which will be affected by the RETS.

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and are applicable throughout these Technical Specifications. Only those definitions applicable to the Radiological Effluent Technical Specifications are included. These terms will be incorporated into the definitions section of the Technical Specifications as a whole.

ACTION

ACTIONS are those additional requirements specified as corollary statements to each specification and shall be part of the specifications.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment as necessary of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is functionally tested.

DOSE EQUIVALENT I-131

The DOSE EQUIVALENT I-131 shall be that concentration of I-131, uCi/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 following is defined equivalent to 1 uCi of I-131 as determined from Table III of TID-14844,"Calculation of Test Reactor Sites": I-132, 28 uCi; I-133, 3.7 uCi; I-134, insignificant; I-135, 12 uCi.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

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.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational or other purposes not associated with plant functions. This category shall not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for the purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCULATION MANUAL

The OFFSITE DOSE CALCULATION MANUAL (ODCM) is a manual which contains the current methodology and parameters to be used to calculate offsite doses resulting from the release of radioactive gaseous and liquid effluents; the methodology to calculate gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints; and, the requirements of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electric power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION

An OPERATIONAL CONDITION shall be any one inclusive combination of mode switch position and average reactor coolant temperature as indicated in Table 1.2.

PROCESS CONTROL PROGRAM

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formula, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71, and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

PURGE - PURGING

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

RATED THERMAL POWER

RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2436 MWT.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee, as defined by Figure 5.1.3-1.

SOLIDIFICATION

SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purpose of protection of individuals from exposure to radiation and radioactive materials OR any area within the SITE BOUNDARY used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

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.

VENTING

VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, 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.

TABLE 1.1

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.
SM	At least once per 16 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Prior to each release.

TABLE 1.2

OPERATIONAL CONDITIONS

OPERATIONAL CONDITIONS	MODE SWITCH POSITIONS	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown	> 212°F
4. COLD SHUTDOWN	Shutdown	≤ 212°F
5. REFUELING*	Refuel**	≤ 212°F

*Reactor vessel head unbolted or removed and fuel in vessel.***
**See Special Test Exception 3.10.3.
***See Special Test Exception 3.10.1.

SECTION 3/4.11-15

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS FOR THE BRUNSWICK STEAM ELECTRIC PLANT

3/4.11 LIQUID EFFLUENTS

3/4.11.1 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.11.1 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.11.1-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2 are not exceeded. The alarm/trip setpoints shall be determined in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.11.1-1

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoints less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, declare the channel inoperative, or change the setpoint so it is acceptably conservative.

b. With less than one radioactive liquid effluent monitoring instrumentation channel in each release pathway OPERABLE, take the ACTION shown in Table 3.11.1-1. Return the instruments to OPERABLE status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.8b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1 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 4.11.1-1.

BASES:

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 and adjusted in accordance with 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. The purpose of tank level indicating devices is to assure the detection and control of leaks that, if not controlled, could potentially result in the transport of radioactive materials to UNRESTRICTED AREAS. "Without delay" implies that the operator, upon determining the LCO is being exceeded, takes the next appropriate action to comply with the specification.

TABLE 3.11.1-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	Instrument	Applicability	Action
a.	Liquid Radwaste Radioactivity Effluent Monitor (Providing alarm and automatic termination of release)	*	110
b.	Liquid Radwaste Effluent Flow Measurement Device	*	111
c.	Service Water Effluent Radioactivity Monitor	*	112
d.	Stabilization Pond Effluent Composite Sampler	**	113
e.	Stabilization Pond Effluent Flow Measuring Device	**	114
f.	Outside Tank Level Indicating Devices--Units 1 and 2 CSTs	*	115
g.	Service Water Effluent from AOG Precooler Radioactivity Monitor	***	112

TABLE NOTATION

* At all times

** During releases via this pathway. [This equipment is to be installed. Prior to installation, appropriate action statements 113 or 114 will be implemented.]

*** At all times once this monitor is installed and after the AOG system becomes operational; however, if the AOG system becomes operational prior to the monitor being installed, then action statement 112 will be implemented. (NOTE: This monitor is to be installed).

TABLE NOTATION (CONTINUED)

ACTION 110 - With less than one channel OPERABLE, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.2.2, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving;

Otherwise suspend release of radioactive effluents via this pathway.

ACTION 111 - With less than one channel OPERABLE, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves or tank level indicators may be used to estimate flow.

ACTION 112 - With less than one channel OPERABLE, effluent releases may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcuries per gram.

ACTION 113 - With the stabilization pond effluent composite sampler not OPERABLE, effluent releases may continue provided that, at least once per day, a grab sample is collected and analyzed for principle gamma emitters as per Table 4.11.2-1. Otherwise, suspend releases via this pathway.

TABLE 3.11.1-1

TABLE NOTATION (continued)

ACTION 114 - With the stabilization pond effluent flow measuring device not OPERABLE, effluent releases via this pathway may continue provided that flow is estimated at least once per day during actual releases. The V-notch weir may be used to estimate flow.

ACTION 115 - With the tank liquid level device not OPERABLE, liquid additions may continue provided the tank liquid level is estimated once per 8 hours during all liquid additions and deletions to and from the tank.

TABLE 4.11.1-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

	Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
a.	Liquid Radwaste Radioactivity Effluent Monitor	D(1)	M	R(2)	Q(3)
b.	Liquid Radwaste Effluent Flow Measurement Device	D(1) (4)	N.A.	R	Q
c.	Service Water Effluent Radioactivity Monitor	D(1)	M	R(2)	Q(3)
d.	Stabilization Pond Effluent Composite Sampler	D(1)	N.A.	R	Q
e.	Stabilization Pond Effluent Flow Measuring Device	D(1)	N.A.	R	Q
f.	Outside Tank Level Indicating Devices-- Units 1 and 2 CSTs	D(5)	N.A.	R	Q
g.	Service Water Effluent From AOG Precooler Radioactivity Monitor	D(1)	M	R(2)	Q

TABLE 4.11.1-1

TABLE NOTATION

1. During releases via this pathway.

2. The initial CHANNEL CALIBRATION 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 shall permit calibrating the system over its intended range of energy and measurement range for subsequent CHANNEL CALIBRATION; sources that have been related to the initial calibration shall be used.

3. The CHANNEL FUNCTIONAL TESTS shall also demonstrate that automatic isolation of this pathway,if applicable, and control room alarm annunciation occurs if any of the following conditions exist:

   a. Instrument indicates measured levels above the alarm/trip setpoint.

   b. High-voltage low.

   c. Instrument indicates a down scale feature.

   d. Instrument controls not set in "operate" mode.

4. The 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. During liquid additions to the tank.

3/4.11.2 LIQUID EFFLUENTS CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.2 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREA (see Figure 5.1.3-1) after dilution in the discharge canal 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} microcuries/ml.

APPLICABILITY: At all times

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.2.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.2-1.

4.11.2.2 The results of radioactivity analyses shall be used in accordance with the methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.2.

BASES:

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS after dilution in the discharge canal 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 in UNRESTRICTED AREAS will not result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC 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.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the Lower Limits of Detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in WASL Procedures Manuals, HASL-300 (revised annually), Currie, L. A. "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

"Without delay" implies that the operator, upon determining the LCD is being exceeded, takes the next appropriate action to comply with the specification.

TABLE 4.11.2-1

BRUNSWICK RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml) a, e
A. Sample Tanks Detergent Drain Tank, and Salt Water Release Tanks (Batch Release)[h]	P Each Batch	P Each Batch	Principal Gamma Emitters[g]	5×10^{-7}[b,d]
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases	1×10^{-5}[d]
	P Each Batch	M Composite[c]	Gross Alpha	1×10^{-7}
			H-3	1×10^{-5}
	P Each Batch	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Stabilization Pond	P Each Release D During Periods of Release[f]	P Each Release D During Periods of Release[f]	Principal Gamma Emitters[g]	5×10^{-7}[b]
C. Service Water	W During System Operation	W During System Operation	Principal Gamma Emitters[g]	5×10^{-7}[b]
D. Circulating Water Pit	P Each Release	P Each Release	Principal Gamma Emitters[g]	5×10^{-7}[b]

TABLE 4.11.2-1 (continued)

TABLE NOTATION

a. The detectability limits for activity analysis are based on technical feasibility limits and on the potential significance in the environment of the quantities released. For some nuclides, Lower Detection Limits may be readily achievable; and when nuclides are measured below the stated limits, they should also be reported.

b. When operational or other limitations preclude specific gamma radionuclide analysis of each batch, gross radioactivity measurements shall be made to estimate the quantity and concentrations of radioactive material released in the batch; and a weekly sample composited from proportional aliquots from each batch released during the week shall be analyzed for principal gamma-emitting radionuclides.

c. 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 that is representative of the liquids released.

d. For certain mixtures of gamma emitters, it may not be possible to measure radionuclides in concentrations near their sensitivity limits when other nuclides are present in the sample in much greater concentrations. Under these circumstances, it will be more appropriate to calculate the concentration of such nuclides using measured ratios with those radionuclides which are routinely identified and measured.

e. The Lower Limit of Detection (LLD) is determined according to the methodology in the ODCM.

TABLE 4.11.2-1 (continued)

TABLE NOTATION

f. The stabilization pond is typically released over a several-day period. The pond is to be sampled and analyzed prior to commencing release. When monitoring instrumentation becomes available and is OPERABLE, daily sampling of the stabilization pond effluent will not be required during release.

g. The principal gamma emitters for which the LLD specifications apply exclusively are 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 that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

h. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling. Once fully operational, the salt water tanks will be included as indicated in Table 4.11.2-1.

3/4.11.3 LIQUID EFFLUENTS DOSE

LIMITING CONDITION FOR OPERATION

3.11.3 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS (see Figure 5.1.3-1) shall be limited:

a. During any calendar quarter to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ, and

b. During any calendar year to $\leq$ 6 mrem to the total body and to $\leq$ 20 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid effluents exceeding any of the limits in Specification 3.11.3, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective action to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.3 Dose Calculations - Cumulative dose contributions from liquid effluents shall be determined in accordance with the Off-site Dose Calculation Manual (ODCM) at least once per 31 days.

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 of 10 CFR Part 50 to assure that releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculations in the ODCM 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 a MEMBER OF THE PUBLIC 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 will be 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 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

The dose or dose commitment to a MEMBER OF THE PUBLIC is based on the 10 CFR Part 50, Appendix I, guideline of:

a. 1.5 mrem to the total body and 5.0 mrem to any organ during any calendar quarter, and

b. 3 mrem to the total body and 10 mrem to any organ during any calendar year,

from radioactive material in liquid effluents from each reactor unit to UNRESTRICTED AREAS.

3/4.11.4 LIQUID RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.4 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site to UNRESTRICTED AREAS (see Figure 5.1.3-1) would exceed 0.12 mrem to the total body or 0.4 mrem to any organ in a 31-day period.

APPLICABILITY: At all times

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the limits in Specification 4.11.4, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystem, and reason for the inoperability.

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

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

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

SURVEILLANCE REQUIREMENTS

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

BASES

The requirement that 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 reasonably achievable." This specification implements the requirements of 10CFR Part 50.36a, General Design Criteria 60 of Appendix A to 10 CFR Part 50 and the design objectives 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.

Mechanical filtration as per system design is considered to be an appropriate component of the liquid radwaste treatment system.

The requirements of 0.12 mrem total body or 0.4 mrem to any organ in a 31-day period is based on 2 reactor units having a shared liquid radwaste treatment system.

3/4.11.5 LIQUID HOLDUP TANKS

Appropriate alternatives to the ACTIONS and Surveillance requirements below can be accepted if they provide reasonable assurance that in the event of an uncontrolled release of the tanks' content, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

LIMITING CONDITION FOR OPERATION

3.11.5 The quantity of radioactive material suspended in solution in each of the following unprotected outdoor tanks shall be limited to less than or equal to the activity indicated below, excluding tritium and dissolved or entrained gases.

	OUTSIDE TANK	CURIE LIMIT
a.	Condensate Storage Tank 1	10 Ci
b.	Condensate Storage Tank 2	10 Ci
c.	Outside Temporary Tank	10 Ci

APPLICABILITY: At all times

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the limit of Specification 3.11.5, without delay suspend all addition of radioactive material to the tank; and within 48 hours reduce the tank's contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.5 The quantity of radioactive material contained in each of the tanks listed in Specification 3.11.5 shall be determined to be within the limit of Specification 3.11.5 by analyzing a representative sample of the tank's contents at least once per seven days when radioactive materials are being added to the tank.

BASES

The tanks listed in this specification include all those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system with the exception of the auxiliary surge tank. The auxiliary surge tank is excluded from this specification because there could be no uncontrolled release of the tank's contents (even in the event of a tank rupture) due to the storm drain collection system. The storm drains empty to a collection basin; the basin is sampled before being pumped to a stabilization pond, and the pond effluents to the intake canal are sampled and released activity is accounted for.

Since the condensate storage tanks have continuous influent and effluent, stratification should not occur. Samples taken from the operating condensate transfer pump(s) vent shall be deemed representative of this system.

3/4.12 GASEOUS EFFLUENTS

3/4.12.1 RADIOACTI E GASEOUS EFFLUENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.12.1.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.12.1-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.12.2 are not exceeded. The setpoints shall be determined in accordance with the methodology as described in the ODCM.

3.12.1.2 The main condenser air ejector monitoring instrumentation channels shown in Table 3.12.1-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.12.6 are not exceeded. The setpoints shall be determined in accordance with the methodology as described in the OCDM.

APPLICABILITY: As shown in Table 3.12.1-1

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specifications 3.12.1.1 or 3.12.1.2 are met, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than one radioactive gaseous effluent or main condenser air ejector monitoring instrumentation channel OPERABLE, take the ACTION shown in Table 3.12.1-1. Return the instruments to OPERABLE status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.8b are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, CHANNEL FUNCTIONAL TEST, and at the frequencies shown in Table 4.12.1-1.

BASES

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 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 and 64 of Appendix A to 10 CFR Part 50.

The main condenser air ejector monitoring instrumentation channels are provided to monitor and control gross radioactivity removed from the main condenser. The alarm/trip setpoint for this monitor shall be calculated in accordance with NRC approved methods to provide reasonable assurance that the potential total body accident dose will not exceed a fraction of the limits specified in 10 CFR Part 100.

"Without delay" implies that the operator, upon determining the LCO is being exceeded, takes the next appropriate action to comply with the specification.

TABLE 3.12.1-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	APPLICABILITY	ACTION
1.	Main Stack Monitoring System		
	a. Noble Gas Activity Monitor	*	123
	b. Iodine Sampler Cartridge	*	127
	c. Particulate Sampler Filter	*	127
	d. System Effluent Flow Rate Measurement Device	*	122
	e. Sampler Flow Rate Measurement Device	*	122
2.	Reactor Building Ventilation Monitoring System		
	a. Noble Gas Activity Monitor	*	123
	b. Iodine Sampler Cartridge	*	127
	c. Particulate Sampler Filter	*	127
	d. System Effluent Flow Rate Measurement Device	*	122
	e. Sampler Flow Rate Measurement Device	*	122

TABLE 3.12.1-1 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	APPLICABILITY	ACTION
3.	Turbine Building Ventilation Monitoring System		
	a. Noble Gas Activity Monitor	*	123
	b. Iodine Sampler Cartridge	*	127
	c. Particulate Sampler Filter	*	127
	d. System Effluent Flow Rate Measurement Device	*	122
	e. Sampler Flow Rate Measurement Device	*	122
4.	Main Condenser Air Ejector Radioactivity Monitor (Prior to Input to Treatment System)		
	a. Noble Gas Activity Monitor - Providing Alarm	***	121
5.	Waste Gas Treatment (Downstream of AOG Treatment System)		
	a. Noble Gas Activity Monitor - Providing Alarm	****	123

TABLE 3.12.1-1 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	APPLICABILITY	ACTION
6.	Waste Gas Treatment System Explosive Gas Monitoring System		
	a. Hydrogen Monitor	**	125

TABLE NOTATION

* At all times.

** During main condenser Augmented Off-Gas Treatment System (AOG) operation.

*** During operation of the main condenser air ejector.

**** At all times once the Augmented Off-Gas Treatment System becomes operational.

Table 3.12.1-1 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATION

ACTION 121 - With less than one main condenser air ejector monitoring instrumentation channel OPERABLE, gases from the main condenser off-gas system may be released to the environment for up to 72 hours provided:

a. The augmented off-gas treatment system (once in operation) is not bypassed, and

b. The main stack effluent noble gas activity monitor is OPERABLE; otherwise, be in at least HOT STANDBY within 12 hours.

ACTION 122 - With less than one channel OPERABLE, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 8 hours.

ACTION 123 - With less than one channel OPERABLE, effluent releases via this pathway may continue provided grab samples are taken at least once per 24 hours and these samples are analyzed for gross noble gas activity within 24 hours.

ACTION 125 - With less than one channel OPERABLE in any operating recombiner train, operation of this waste treatment system may continue provided grab samples from the affected train are collected at least once per 24 hours and analyzed within the following 4 hours and proper function of the recombiner is assured by monitoring recombiner temperature in accordance with approved procedures.

Table 3.12.1-1 (continued)

TABLE NOTATION

ACTION 127 - With less than one channel OPERABLE, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment and analyzed as required in Table 4.12.2-1.

TABLE 4.12.1-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
1. Main Stack Monitoring System				
a. Noble Gas Activity Monitor	D(1)	M	R(2)	Q(3)
b. Iodine Sampler Cartridge	W(1)	N.A.	N.A.	N.A.
c. Particulate Sampler Filter	W(1)	N.A.	N.A.	N.A.
d. System Effluent Flow Rate Measurement Device	D(1)	N.A.	R	Q
e. Sampler Flow Rate Measurement Device	D(1)	N.A.	R	Q
2. Reactor Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	D(1)	M	R(2)	Q(3)
b. Iodine Sampler Cartridge	W(1)	N.A.	N.A.	N.A.
c. Particulate Sampler Filter	W(1)	N.A.	N.A.	N.A.

TABLE 4.12.1-1 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
d. System Effluent Flow Rate Measurement Device	D(1)	N.A.	R	Q
e. Sampler Flow Rate Measurement Device	D(1)	N.A.	R	Q
3. Turbine Building Ventilation Monitoring System				
a. Noble Gas Activity Monitor	D(1)	M	R(2)	Q(3)
b. Iodine Sampler Cartridge	W(1)	N.A.	N.A.	N.A.
c. Particulate Sampler Filter	W(1)	N.A.	N.A.	N.A.
d. System Effluent Flow Rate Measurement Device	D(1)	N.A.	R	Q
e. Sampler Flow Rate Measurement Device	D(1)	N.A.	R	Q

TABLE 4.12.1-1 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

Instrument	Channel Check	Source Check	Channel Calibration	Channel Functional Test
4. Main Condenser Air Ejector Radioactivity Monitor (Prior to Input to Treatment System)				
a. Noble Gas Activity Monitor - Providing Alarm	D(1)	M	R(2)	R(3)
5. Waste Gas Treatment (Downstream of AOG Treatment System(5)				
a. Noble Gas Activity Monitor - Providing Alarm	P	P	R(2)	Q
6. Waste Gas Treatment System Explosive Gas Monitoring System(5)				
a. Hydrogen Monitor	D	N.A.	Q(4)	M

TABLE 4.12.1-1 (continued)

TABLE NOTATION

1. During releases via this pathway.

2. The initial CHANNEL CALIBRATION 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 shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

3. The CHANNEL FUNCTIONAL TESTS shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

   a. Instrument indicates measured levels above the alarm/trip setpoint.

   b. High-voltage low.

   c. Instrument indicates a down scale failure.

   d. Instrument not set in "operate" mode.

4. The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

   a. One volume percent hydrogen, balance nitrogen, and

   b. Four volume percent hydrogen, balance nitrogen.

5. Instrumentation for this system is only applicable once the Augumented Off-Gas Treatment System becomes fully operational at the Brunswick Steam Electric Plant.

3/4.12.2 GASEOUS EFFLUENTS DOSE RATE

LIMITING CONDITION FOR OPERATION

3.12.2 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) shall be limited to the following:

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

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

APPLICABILITY: At all times

ACTION:

With the dose rate(s) exceeding the above limits, without delay, restore the release rate to within the above limit(s), and provide prompt notification to the Commission pursuant to Specification 6.9.1.7.

SURVEILLANCE REQUIREMENTS

4.12.2.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology as described in the ODCM.

4.12.2.2 The dose rate due to Iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology as described in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.12.2-1.

BASES

This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose rate limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC, 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 MEMBERS OF THE PUBLIC 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 SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are apportioned equally among the units sharing that system.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the Lower Limits of Detection (LLDs).

Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

"Without delay" implies that the operator, upon determining the LCO is being exceeded, takes the next appropriate action to comply with the specification.

TABLE 4.12.2-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Drywell Purge	P Each Purge Grab Samples	P Each Purge	Principal Gamma Emitters[b]	1×10^{-4}
B. Environmental Release Points - Main Stack, Unit 1 & Unit 2 Reactor Building Vents, Unit 1 & Unit 2 Turbine Building Vents, Hot Shop	M[c, d] Grab Sample	M[c]	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[e]	W[f, g] Charcoal Sample	I-131	1×10^{-12}
	Continuous[e]	W[f, g] Particulate Sample	Principle Gamma Emitter[b] (I-131, others)	1×10^{-11}
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[e]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.12.2-1 (continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATION

a. The Lower Limit of Detection (LLD) is determined in accordance with the methodology as presented in the ODCM.

b. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133 m, Xe-135, Xe-135 m, 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 that are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

c. With a THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER within one hour or following shutdown or start-up, sampling and analyses shall also be performed unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

d. If during refueling, the tritium concentration in the fuel pool water exceeds 2×10^{-4} μCi/ml, 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. Fuel pool water will be sampled at least once per 7 days during refueling.

e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.12.2, 3.12.3, and 3.12.4.

f. Samples cartridges/filters shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler).

g. Sampling shall be performed at least once per 24 hours for at least 7 days following each shutdown, start-up, or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in 1 hour, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

3/4.12.3 GASEOUS EFFLUENTS DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.12.3 The air dose due to noble gases released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter, to $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation;

b. During any calendar year, to $\leq$ 20 mrad for gamma radiation and $\leq$ 40 mrad for beta radiation.

APPLICABILITY: At all times

ACTIONS:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding Limiting Condition For Operation 3.12.3, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases, and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.12.3 Dose Calculations - Cumulative dose contributions for noble gases for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

BASES

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 materials 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 is to be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC 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 will be 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 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 and beyond the SITE BOUNDARY will be based upon the historical annual average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

3/4.12.4 GASEOUS EFFLUENTS DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.12.4 The dose to a MEMBER OF THE PUBLIC from Iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1) shall be limited to the following:

a. During any calendar quarter, less than or equal to 15 mrems to any organ; and

b. During any calendar year, less than or equal to 30 mrems to any organ.

APPLICABILITY: At all times

ACTION:

a. With the calculated dose from the release of Iodine-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.12.4 Dose Calculations - Cumulative dose contributions for the current calendar quarter and current calendar year for I-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the ODCM at least once per 31 days.

BASES

This specification is provided to implement the requirements of Section 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, implements 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 a MEMBER OF THE PUBLIC 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 required to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating 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 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 specification for Iodine-131, tritium, and radioactive material in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. 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.

3/4.12.5 GASEOUS RADWASTE TREATMENT/VENTILATION EXHAUST TREATMENT

LIMITING CONDITION FOR OPERATION

3.12.5 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 effluent releases, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1), would exceed 0.4 mrad for gamma radiation and 0.8 mrad for beta radiation over 31 days. 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 to areas at and beyond the SITE BOUNDARY (see Figure 5.1.3-1), would exceed 0.6 mrem to any organ over 31 days.

APPLICABILITY: At all times

ACTION:

a. With gaseous waste being discharged without treatment and in excess of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability;

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status; and

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

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

SURVEILLANCE REQUIREMENTS

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

BASES

This requirement provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as 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 the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use 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.

Until such time as the Augmented Off-Gas System becomes operational at the Brunswick Steam Electric Plant, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute off-gas holdup line including stack filtration; and the only VENTILATION EXHAUST TREATMENT SYSTEMS shall be those installed for the Turbine Building Ventilation and the Standby Gas Treatment System.

3/4.12.6 MAIN CONDENSER AIR EJECTOR RADIOACTIVITY RELEASE RATE

LIMITING CONDITION FOR OPERATION

3.12.6 The gross radioactivity (beta and/or gamma) rate of noble gases measured at the main condenser air ejector shall be limited to ensure that the total body exposure to an individual in the UNRESTRICTED AREA shall not exceed 2.5 rem in one hour's time.

APPLICABILITY: At all times

ACTION:

With the gross radioactivity (beta and/or gamma) rate of noble gases at the main condenser air ejector exceeding the above limit, restore the gross radioactivity rate to within its limit within 72 hours or be in at least HOT STANDBY within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.12.6.1 The main condenser air ejector radioactivity release rate shall be determined in accordance with the methodology as described in the ODCM.

4.12.6.2 The gross radioactivity (beta and/or gamma) rate of noble gases from the main condenser air ejector shall be determined to be within the above limit at the following frequencies by performing an isotopic analysis of a representative sample of gases taken at the discharge (prior to dilution and/or discharge) of the main condenser air ejector:

a. At least once per 31 days.

b. Within 72 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, or greater than 50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual in the UNRESTRICTED AREA will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.12.7 EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.12.7 The concentration of hydrogen in the waste gas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times (see Bases)

ACTION:

a. With the concentration of hydrogen in the waste gas treatment system exceeding the limit, restore the concentration to within the limit within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.12.7 The concentration of hydrogen in the waste gas treatment system shall be determined to be within the above limit by continuously monitoring the waste gases in the waste gas treatment system with the hydrogen monitors required OPERABLE by Table 3.12.1-1 of Specification 3.12.1.1.

BASES

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen. Maintaining the concentration of hydrogen below the 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.

This specification will become applicable when the Augmented Off-Gas System becomes fully operational at the Brunswick Steam Electric Plant. There is no requirement for hydrogen monitors on the 30-minute waste gas holdup line which will serve in the interim.

3/4.12.8 DRYWELL PURGES (MARK I CONTAINMENT)

LIMITING CONDITION FOR OPERATION

3.12.8 The drywell shall be purged through the Standby Gas Treatment System or released to the environment at a rate in conformance with Specification 3.12.2.

APPLICABILITY: Whenever the drywell is vented or purged.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all VENTING or PURGING of the drywell.

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

SURVEILLANCE REQUIREMENTS

4.12.8 A sample analysis, as defined in Table 4.12.2-1, shall be performed prior to each drywell PURGE.

BASES

This specification provides reasonable assurance that releases from drywell PURGING operations will not exceed the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS.

3/4.13 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.13 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.8b are not applicable.

SURVEILLANCE REQUIREMENTS

4.13 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection of testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.15, to assure SOLIDIFICATION of subsequent batches of waste.

BASES

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included 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.

3/4.14 40 CFR PART 190

3/4.14.1 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.3a, 3.11.3b, 3.12.3a, 3.12.3b, 3.12.4a, or 3.12.4b, calculations should be made to determine whether the above limits of Specification 3.14.1 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits; and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report

shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUREMENTS

4.14.1 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.3, 4.12.3, and 4.12.4, and in accordance with the ODCM. Only uranium fuel cycle sources within five miles of the Brunswick Steam Electric Plant will be considered.

BASES

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER 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 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected) in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c is considered to be a timely request and ful-

fills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3/4.11 and 3/4.12. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.15 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.15.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.15.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.15.11-1.

APPLICABILITY: At all times

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.15.1-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampliмg medium at a specific location exceeding the reporting levels of Table 3.15.1-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective action to be taken to reduce radioactive effluents so that the potential annual dose to a member of the public is less than the calendar year limits of Specifications 3.11.3, 3.12.2, and 3.12.4. When more than one of the radionuclides in Table 3.15.1-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 6.9-1 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.3, 3.12.3, and 3.12.4. 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.

c. With milk or fresh leafy vegetables unavailable from one or more of the sample locations required by Table 3.15.1-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program and ODCM. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the cause of unavailability of samples; and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report, and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.15.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.15.1-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 3.15.1-1 and the detection capabilities required by Table 4.15.1-1.

BASES

The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluents monitoring program by verifying that the measurable concentrations of radioactive materials are not higher than expected on the basis of effluent measurements and the modeling of the environmental exposure pathways.

The required detection capabilities for environmental sample analysis are tabulated in terms of the Lower Limits of Detection (LLDs). The LLDs required by Table 4.15.1-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as 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.

Detailed discussion of the LLD and other detection limits can be found in HASL Procedure Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determinator Application to Radiochemsitry" Anal. Chem 40, 586-93 (1968), and Hartwell, L. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

Groundwater is not monitored by this specification because plant liquid effluents are not tapped as a source for drinking or irrigation purposes.

The following notes provide the specific bases for the Radiological Monitoring Program summarized in Table 3.15.1-1.

1. Direct Radiation - At least 40 routine monitoring stations with two or more dosimeters or one instrument for measuring and recording dose rate continuously are placed as described in the ODCM. The stations are

located to provide an inner ring of stations in each sector located as near the SITE BOUNDARY as is reasonably accessible and practical so as to estimate the highest possible dose to a MEMBERS(s) OF THE PUBLIC. An additional outer ring of stations is also included at distances of 8 km or greater, selected in each sector on the basis of accessibility. Sectors extending over open water are omitted. Additional stations are added to represent nearby populated areas such as a religious assembly area, recreational area, and permanent residences. An additional 3 stations are maintained at distances greater than 16 km to serve as control stations.

2. Airborne - 3 samples are taken at stations located as near to the SITE BOUNDARY as is reasonably accessible. In addition, one sampling station is established in a nearby community and one is located to provide control data.

3. Waterborne - The surface water at BSEP is brackish, estuary water that is discharged into the Atlantic Ocean. This water is not used as water for any kind of drinking or other human consumption nor is it used to irrigate consumable foods but only for recreational activities; therefore, one upstream and one downstream sampling location is established to estimate and monitor plant effect via this pathway.

4. Ingestion

   a. Milk - Samples from milking animals within a 5-km distance are taken. Currently only one such station of a single animal exists. Additions and deletions of stations are accomplished in accordance with Specification 3.15.2. One station of greater than 16 km from the plant is in a low D/Q sector is collected to provide a control data for comparison purposes.

   b. Fish and Invertebrates - One sample is taken representative of consumable species in the vicinity of the discharge point. One sample of similarly consumable species are located in areas upstream or otherwise not influenced by plant discharges.

c. Food Products - Samples of broadleaf, edible vegetation grown in two (2) different sectors of historically higher D/Q values at the SITE BOUNDARY are collected as seasonally available as per Table 3.15.1-1. One additional sample of similar edible vegetation grown at a distance of about 10 miles from the plant is also taken to provide control data when it is seasonally available as per Table 3.15.1-1.

TABLE 3.15.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR PATHWAY	NUMBER OF SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation	Locations 1-40. At each location with 2 or more dosimeters or one or more instruments for continuously measuring and recording dose rate.	Q	Gamma Dose - Q
2. Airborne - Radioiodine and Particulate	Locations 41-45	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent.	Radioiodine Cannister - W Particulate sampler - Analyze for gross beta radioactivity > 24 hours following filter change. Perform gamma isotopic analysis on each sample when gross beta activity is > 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
3. Waterborne a. Surface	Locations 46-47	Composite[b] sample collection - M	Gamma Isopic Analysis - M Tritium - Q Analysis
b. Sediment from shoreline	Location 48	SA	Gamma Isotopic SA

TABLE 3.15.1-1 (continued)

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion a. Milk[c]	Locations 49, 50	With animals on pasture-SM At other times - M	Gamma isotopic (animals on pasture) SM I-131 analyses (animals on pasture) SM Gamma Isotopic (other times) M I-131 analysis (other times) M
b. Fish and Invertebrates	Postions 51, 52	When in season - SA	Gamma isotopic on edible portions - SA
c. Food Products[c]	Positions 53-56	When available - M	Gamma isotopic - M I-131 - M

TABLE 3.15.1-1 (continued)

TABLE NOTATION

a. Actual locations (distance and direction) from the site are provided for all sample locations in a table and a figure(s) in the ODCM.

b. Composite samples shall be collected with equipment that is capable of collecting an aliquot at time intervals that are short (e.g., once per 6 hours) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

c. When less than three (3) milking animals are available for testing within an 8-km distance and when the dose to a MEMBER OF THE PUBLIC at 3 km in the highest X/Q sector, calculated in accordance with the ODCM for a milk pathway is greater than 1 mrem per year, sampling of food products shall be performed as indicated in Table 3.15.1-1, 4.c, in lieu of milk sampling.

TABLE 3.15.1-2

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)	Food Products (pCi/kg, wet)
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	

TABLE 4.15.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

LOWER LIMIT OF DETECTION (LLD)

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01	-	-	-	-
H_3	2000	-	-	-	-	-
Mn-54	15	-	130	-	-	-
Fe-59	30	-	260	-	-	-
Co-58, 60	15	-	130	-	-	-
Zn-65	30	-	260	-	-	-
Zr-Nb-95	15	-	-	-	-	-
I-131	1*	0.07	-	1	60	-
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15	-	-	15	-	-

Note: This list does not mean that these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported. The LLD is determined according to methodology in the ODCM.

* LLD for drinking water

3/4.15.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.15.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest resident, and the nearest garden of greater than 50 m^2 (500 ft^2) producing broadleaf vegetation. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify within a distance of 5 km (3 miles) the location in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m^2 producing broadleaf vegetation.)

Broadleaf vegetable sampling of at least 3 different kinds of vegetation may be performed at the site boundary in each of 2 different direction sectors with the highest D/Qs in lieu of the garden census. Specifications for broadleaf vegetation sampling in Table 3.15.1-1(4c) shall be followed, including analysis of control samples.

APPLICABILITY: At all times

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.12.4, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.9.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.15.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the

central station location, having the lowest calculated dose or dose commitment(s) (via this same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.9, identify the new location(s) in the next Semiannual Effluent Release Report; and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.15.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The result of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

BASES

This specification is provided to ensure that changes in the use of area at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of the census. The best information from door-to-door surveys, aerial surveys, or consulting with local agricultural 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 50 m^2 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/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine the minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broadleaf vegetation (i.e., similar to lettuce and cabbage; and (2) a vegetation yield of 2 kg/m^2.

3/4.15.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.15.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

APPLICABILITY: At all times

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

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

SURVEILLANCE REQUIREMENTS

8.5.3 A summary of the results, obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM, shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

BASES

The requirement for participation in the 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 the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

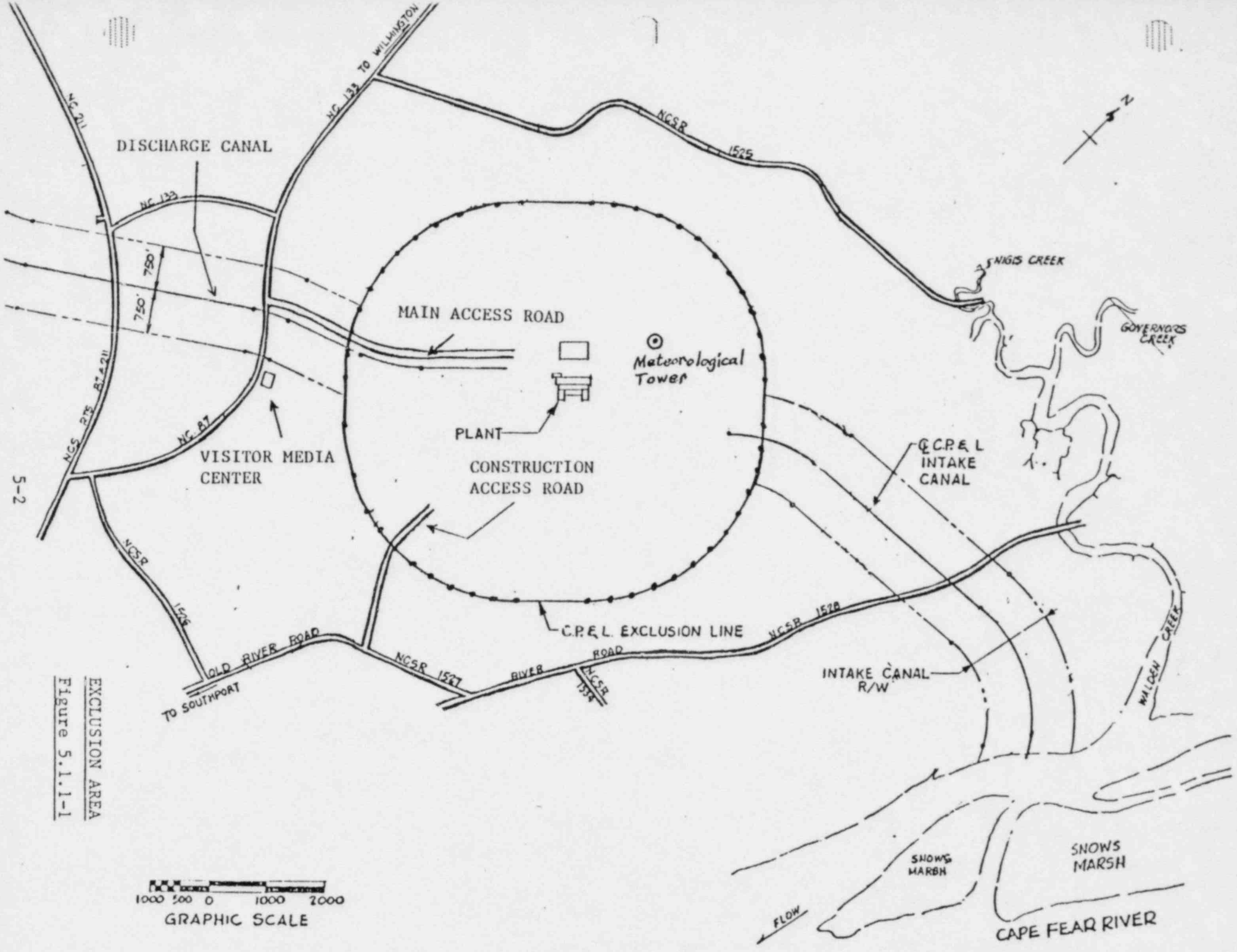
5.1.1 The exclusion area shall be as shown in Figure 5.1.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1.2-1.

SITE BOUNDARY

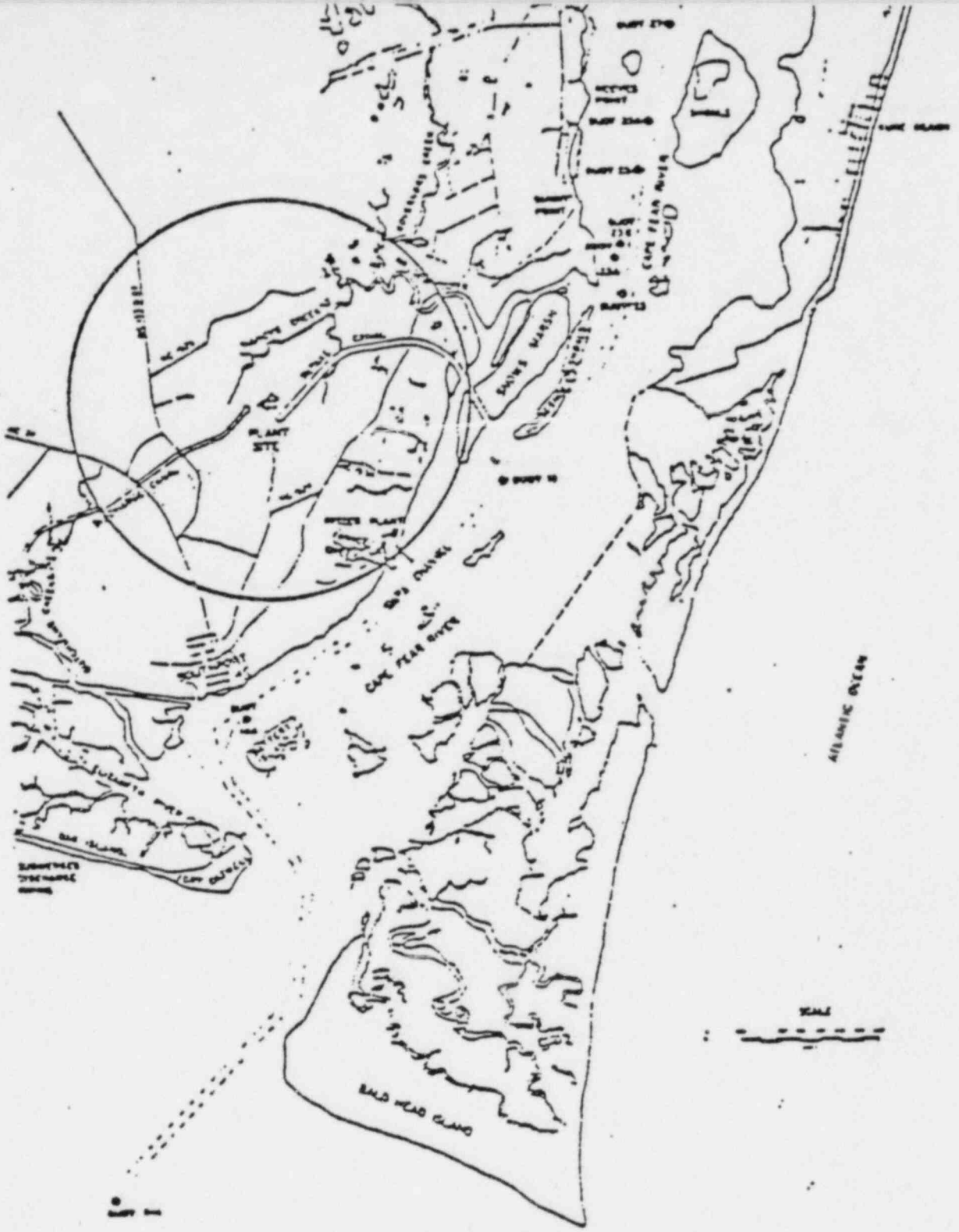
5.1.3 The SITE BOUNDARY shall be as shown in Figure 5.1.3-1. For the purpose of effluent release calculations, the boundary for atmospheric releases is the SITE BOUNDARY and the boundary for liquid releases is the SITE BOUNDARY prior to dilution in the Atlantic Ocean.



5-2

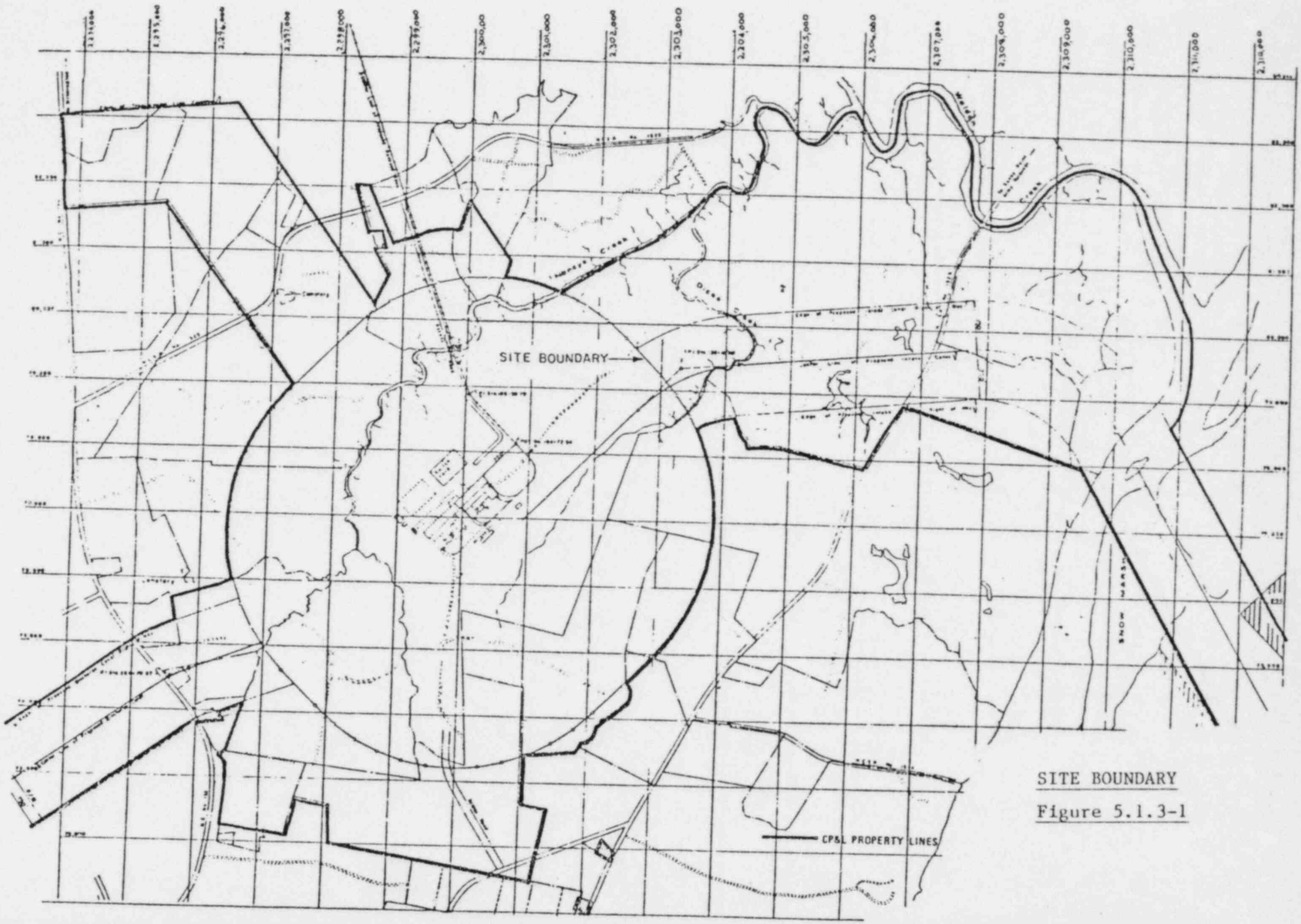
EXCLUSION AREA

Figure 5.1.1-1



LOW POPULATION ZONE

FIGURE 5.1.2-1



SITE BOUNDARY

Figure 5.1.3-1

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2.1-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figures 6.2.2-1 and 6.2.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1.

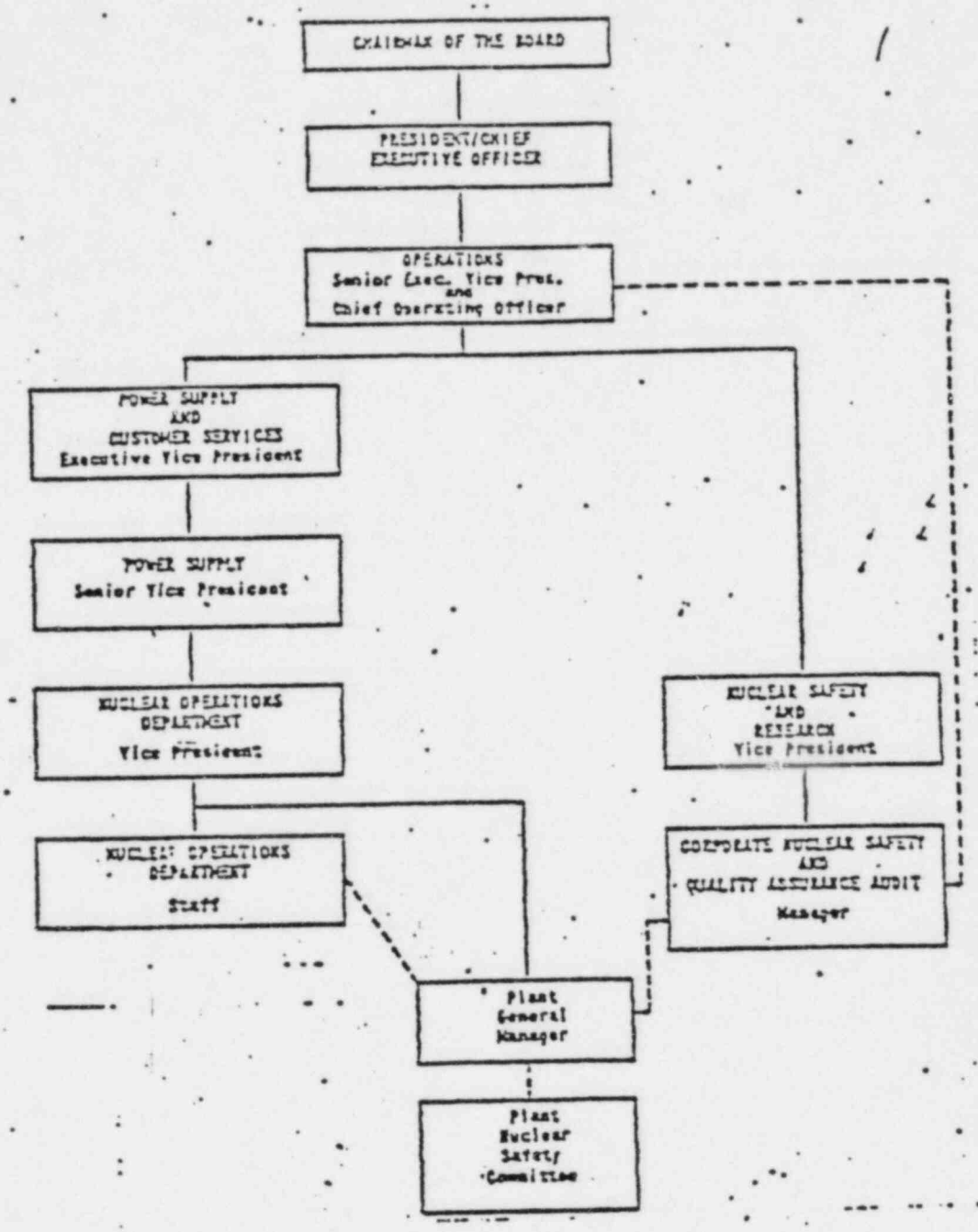
b. At least one licensed Operator shall be in the control room for each reactor containing fuel.

c. At least two licensed Operators shall be present in the control room for each reactor in the process of start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified to implement radiation protection procedures shall be on site when fuel is in either reactor.

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

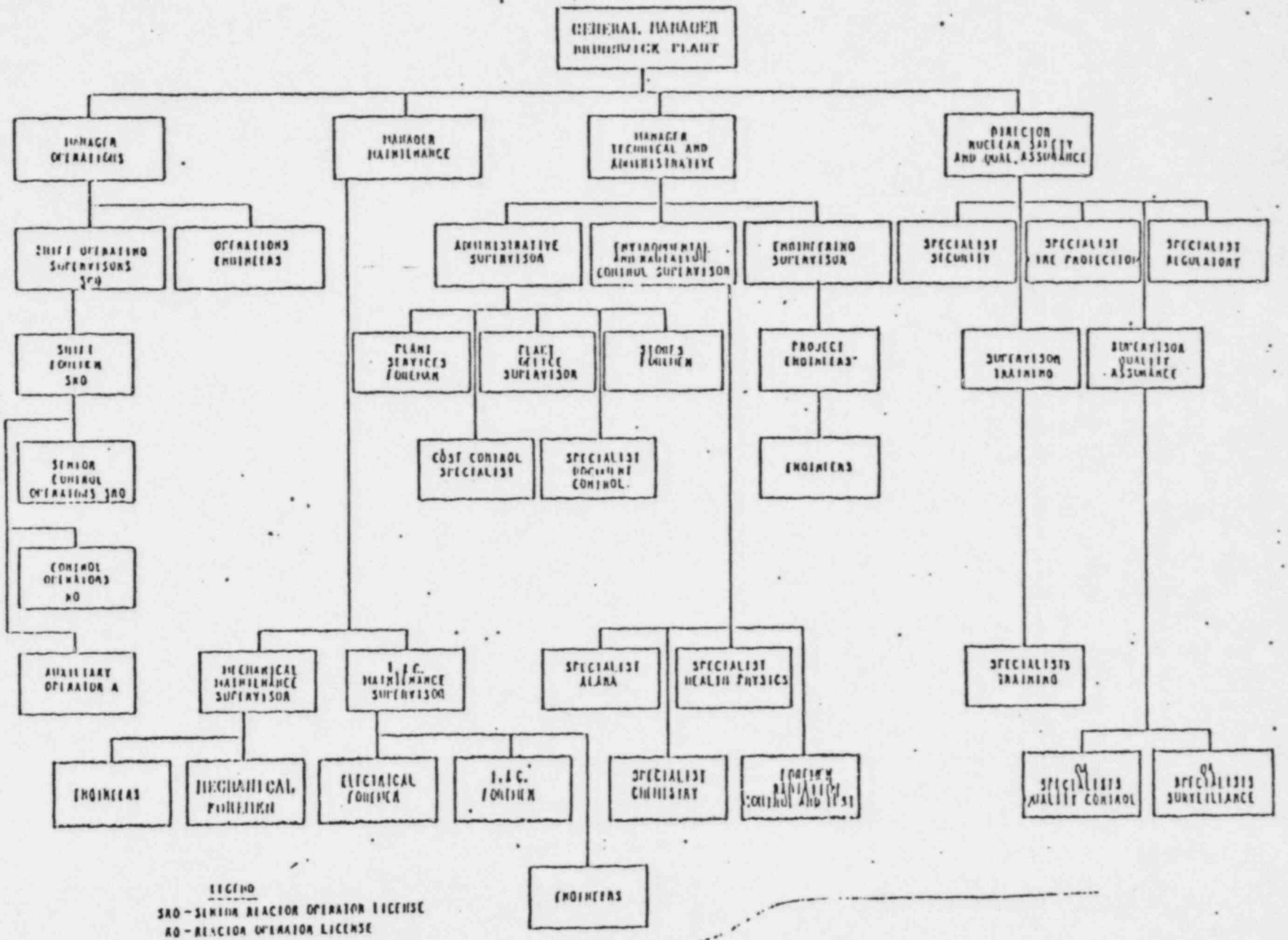
f. A Fire Brigade of at least five members shall be maintained onsite at all times. The Fire Brigade shall not include the minimum shift crew shown in Table 6.2.2-1 or any personnel required for other essential functions during a fire emergency.



*Responsible for performance and monitoring of Fire Protection Program.

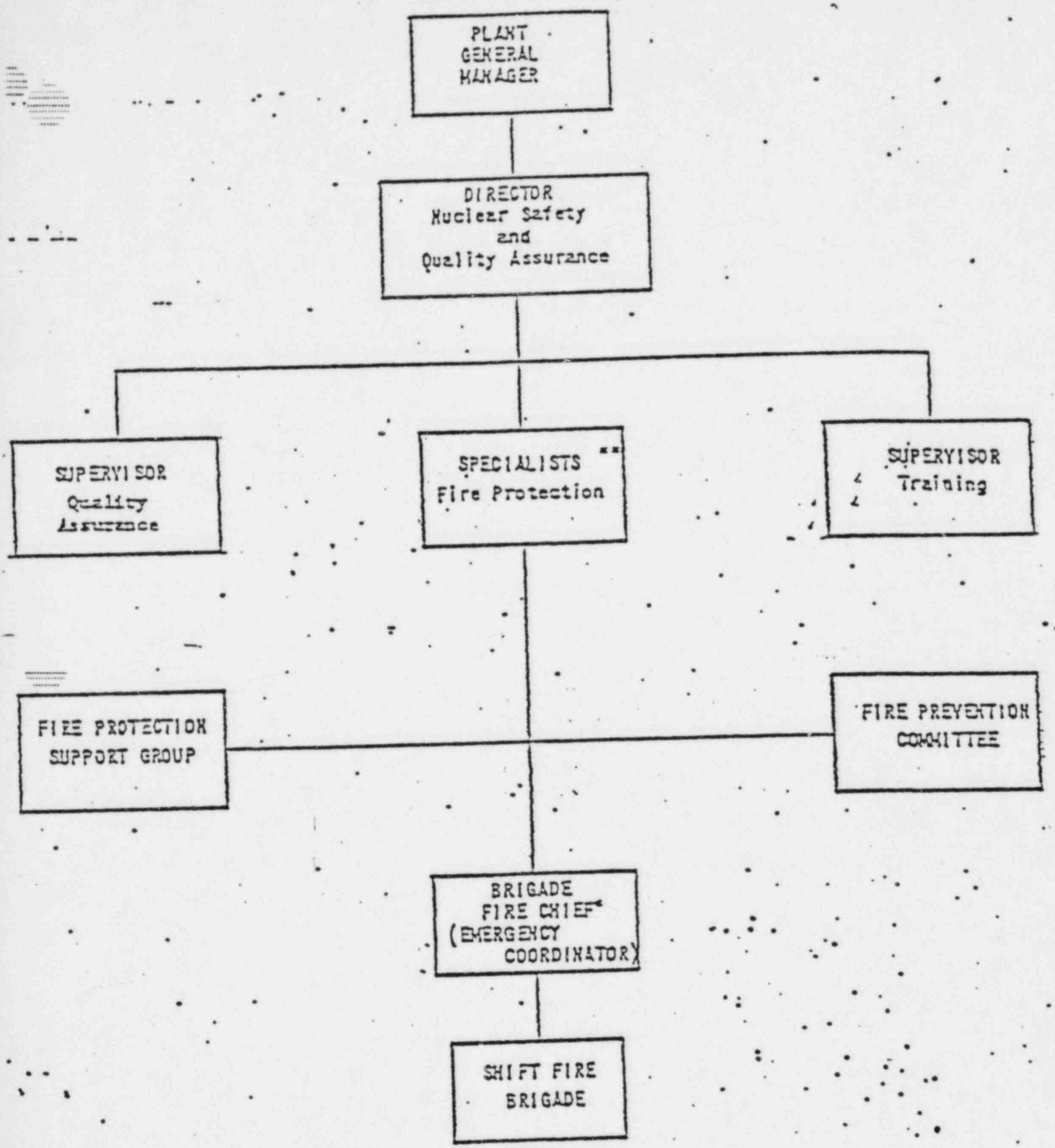
MANAGEMENT ORGANIZATION CHART

FIGURE 6.2-1



FACILITY ORGANIZATION

Figure 6.2.2-1



PLANT FIRE PROTECTION ORGANIZATION

*Number of Brigade Fire Chiefs varies with shift organization.
**One Engineer is assigned the duties of the plant fire chief.

FIGURE 6.2.2-2

TABLE 6.2.2-1

MINIMUM SHIFT CREW COMPOSITION#

Condition of Unit 1 - Unit 2 in CONDITION 1, 2 or 3

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	2*
OL**	3	2
Non-Licensed	4	3

Condition of Unit 1 - Unit 2 in CONDITION 4 or 5

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL**	2	1*
OL**	2	2
Non-Licensed	3	3

Condition of Unit 1 - No Fuel in Unit 2

LICENSE CATEGORY	APPLICABLE OPERATIONAL CONDITIONS	
	1, 2, 3	4 & 5
SOL	1	1*
OL	2	1
Non-Licensed	2	1

* Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

**Assumes each individual is licensed on both plants.

Shift crew composition, including an individual qualified in radiation protection procedures, may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for the Environmental and Radiation Control Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Plant Fire Chief and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.1.1 The PNSC shall function to advise the General Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PNSC shall be composed of the:

Chairman:	Plant General Manager
Vice Chairman:	Operations Manager, Maintenance Manager, Technical - Administrative Manager or Director-Nuclear Safety and QA
Secretary:	Administrative Supervisor
Member:	Maintenance Supervisor (I&C)
Member:	Maintenance Supervisor (Mechanical)
Member:	Engineering Supervisor
Member:	Environmental and Radiation Control Supervisor
Member:	Quality Assurance Supervisor
Member:	Shift Operating Supervisors
Member:	Training Supervisor

ALTERNATES

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

MEETING FREQUENCY

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

QUORUM

6.5.1.5 A quorum of the PNSC shall consist of the Chairman or Vice Chairman and three members including alternates.

RESPONSIBILITIES

6.5.1.6 The PNSC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the General Manager to affect nuclear safety.

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

c. Review of all proposed changes to Technical Specifications.

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

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Operations and to the Manager - Corporate Nuclear Safety and Quality Assurance Audit.

f. Review of all events requiring 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews, investigations and reports thereon as requested by the Manager - Corporate Nuclear Safety and Quality Assurance Audit.

i. Review of the Plant Security Plan and implementing procedures.

j. Review of the Emergency Plan and implementing procedures.

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President-Nuclear Operations and to the Corporate Nuclear Safety Unit.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

AUTHORITY

6.5.1.7 The PNSC shall:

a. Recommend to the General Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

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

c. Provide written notification within 24 hours to the Vice President - Nuclear Operations and the Manager - Corporate Nuclear Safety and Quality Assurance Audit of disagreement between the PNSC and the General Manager; however, the General Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PNSC shall maintain written minutes of each meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility and authority provisions of these technical specifications, and copies shall be provided to the Vice President - Nuclear Operations and to the Manager - Corporate Nuclear Safety and Quality Assurance Audit.

6.5.2 CORPORATE NUCLEAR SAFETY AND QUALITY ASSURANCE AUDIT SECTION (CNS & G

RESPONSIBILITY

6.5.2.1 The Manager - Corporate Nuclear Safety and Quality Assurance Audit, under the Vice President - Nuclear Safety and Research, is charged with the overall responsibility for administering the independent off-site review and quality assurance audit programs as follows:

a. Approves selection of the individuals to conduct off-site safety reviews and quality assurance audits.

b. Has access to the plant operating records and operating personnel in performing the independent reviews and quality assurance audits

c. Prepares and retains written records of review and audits.

d. Assures independent safety reviews are conducted on all items required by Section 6.5.3.3 and quality assurance audits cover all items included in Section 6.5.4.1.

e. Distributes reports, records of PNSC meetings, and other records to the appropriate managers and individuals assigned to conduct the off-site safety reviews and quality assurance audits.

6.5.3 CORPORATE NUCLEAR SAFETY UNIT (CNSU)

FUNCTION

6.5.3.1 The Corporate Nuclear Safety Unit of the Corporate Nuclear Safety and Quality Assurance Audit Section shall provide independent off-site review of significant plant changes, tests, and procedures; verify that reportable occurrences are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

PERSONNEL

6.5.3.2

a. Personnel assigned responsibility for independent reviews shall be specified in technical disciplines, and shall collectively have the experience and competence required to review problems in the following areas:

1. Nuclear power plant operations
2. Nuclear engineering
3. Chemistry and radiochemistry
4. Metallurgy
5. Instrumentation and control
6. Radiological safety
7. Mechanical and electrical engineering
8. Administrative controls
9. Seismic and environmental
10. Quality assurance practices

b. The following minimum experience requirements shall be established for those persons involved in the independent off-site safety review program:

1. Manager of CNS and QAAS - Bachelor of Science in engineering or related field and ten (10) years related experience including five (5) years involvement with operation and/or design of nuclear power plants.

2. Reviewers - Bachelor of Science in engineering or related field or equivalent and five (5) years related experience including three (3) years involvement with operation and/or design of nuclear power plants.

PERSONNEL (Continued)

c. An individual may possess competence in more than one specialty area. If sufficient expertise is not available within the Corporate Nuclear Safety Unit, competent individuals from other Carolina Power and Light Company organizations or outside consultants shall be utilized in performing independent off-site reviews and investigations.

d. At least three persons, qualified as discussed in Specification 6.5.2.3.b, shall review each item submitted under the requirements of Section 6.5.3.3.

e. Independent safety reviews shall be performed by personnel not directly involved with the activity or responsible for the activity.

SUBJECTS REQUIRING INDEPENDENT REVIEW

6.5.3.3 The following subjects shall be reviewed by the Corporate Nuclear Safety Unit:

a. Written safety evaluations of changes in the facility as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests, or experiments did not involve a change in the technical specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2).

b. Proposed changes in procedures, proposed changes in the facility, or proposed tests or experiments, any of which involves a change in the Technical Specifications or an unreviewed safety question pursuant to 10 CFR 50.59(c). Matters of this kind shall be referred to the Corporate Nuclear Safety Unit by the Plant Nuclear Safety Committee following its review, or by other functional organizational units within Carolina Power & Light Company prior to implementation.

c. Proposed changes to the Technical Specifications or this operating license.

SUBJECTS REQUIRING INDEPENDENT REVIEW (Continued)

d. Violations, deviations and reportable events, which require reporting to the NRC within 24 hours, and as defined in the plant technical specifications such as:

1. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements or internal procedures or instructions having safety significance; and

2. Significant operating abnormalities or deviations from normal or expected performance of plant safety-related structures, systems, or components.

Review of events covered under this paragraph shall include the results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Any other matter involving safe operation of the nuclear power plant which the Manager - Corporate Nuclear Safety and Quality Assurance Audit Section deems appropriate for consideration, or which is referred to the Manager - Corporate Nuclear Safety and Quality Assurance Audit Section by the onsite operating organization or by other fuctional organizational units within Carolina Power and Light Company.

f. Reports and meeting minutes of the PNSC.

FOLLOW-UP ACTION

6.5.3.4 Results of Corporate Nuclear Safety (CNS) reviews, including recommendations and concerns shall be documented.

a. Copies of the documented review shall be retained in the Corporate Nuclear Safety and Quality Assurance Audit Section files.

b. Recommendations and concerns shall be submitted to the Vice President - Nuclear Operations within 14 days of determination.

c. A summation of Corporate Nuclear Safety recommendations and concerns shall be submitted to the Chairman/Chief Executive Officer; Senior Executive Vice President and Chief Operating Officer; Executive Vice President - Power Supply and Customer Services; Senior Vice President - Power Supply; Vice President-Nuclear Safety and Research; Plant General Manager; and others, as appropriate on at least a bi-monthly frequency.

6.5.3.5 The Corporate Nuclear Safety Unit review program shall be conducted in accordance with written, approved procedures.

6.5.4 OPERATION AND MAINTENANCE UNIT (OMU)

FUNCTION

6.5.4.1 The Operation and Maintenance Unit of the Corporate Nuclear Safety and Quality Assurance Audit Section shall perform audits of plant activities. These audits shall encompass:

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

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

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

d. The verification of compliance and implementation of the requirements of the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

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

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

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

h. Any other area of facility operation considered appropriate by the Corporate Quality Assurance Audit Operation and Maintenance Unit, the Executive Vice President - Power Supply and Customer Service, or the Senior Vice President - Power Supply.

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

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

k. The PROCESS CONTROL PROGRAM and implementing procedures at least once per 24 months.

l. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.2, Revision 1, June 1974, and Regulatory Guide 4.1, Revision 1, April 1975, at least once per 12 months.

PERSONNEL

6.5.4.2

a. Audit personnel shall be independent of the area audited. Selection for auditing assignments is based on experience or training which establishes that their qualifications are commensurate with the complexity or special nature of the activities to be audited. In selecting auditing personnel, consideration shall be given to special abilities, specialized technical training, prior pertinent experience, personal characteristics, and education.

b. Qualified outside consultants or other individuals independent from those personnel directly involved in plant operation, but within the Operations Group, shall be used to augment the audit teams when necessary.

REPORTS

6.5.4.3 Results of audit are approved by the Manager - Corporate Nuclear Safety and Quality Assurance Audit Section and transmitted directly to the Company President/Chief Executive Officer, the Senior Executive Vice President and Chief Operating Officer, the Executive Vice President - Power Supply and Customer Services, the Senior Vice President - Power Supply, and the Vice President - Nuclear Safety and Research, and others, as appropriate within 30 days after the completion of the audit.

6.5.4.4 The Corporate Quality Assurance Audit Program shall be conducted in accordance with written, approved procedures.

6.5.5 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

6.5.5.1 An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing an outside fire protection firm.

6.5.5.2 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.

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 PNSC and submitted to Manager - Corporate Nuclear Safety and Quality Assurance Audit and the Vice President - 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 SHUTDOWN within two hours.

b. The Safety Limit violation shall be reported to the Commission, the Vice President - Nuclear Operations and to the Manager - Corporate Nuclear Safety and Quality Assurance Audit within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PNSC. 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 Manager - Corporate Nuclear Safety and Quality Assurance Audit and the Vice President - 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. The onsite portion of the radiological environmental monitoring program implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. PROCESS CONTROL PROGRAM implementation.

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

PROCEDURES (Continued)

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PNSC and approved by the General Manager prior to implementation and reviewed periodically by the PNSC as set forth in administrative procedures.

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 members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the Brunswick Plant.

c. The change is documented, reviewed by the PNSC and approved by the General Manager 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 REPORT

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

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

6.9.1.2 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,

STARTUP REPORT (Continued)

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/]

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[1]

6.9.1.3 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.

6.9.1.4 These reports shall include:

1. Summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment;

2. The results of land use census required by Specification 3.15.2.

3. The results of analysis of all radiological environmental samples and of all measurements taken during the period pursuant to the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of Table 6.9-1. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data should be submitted as soon as possible in a supplementary report;

4. A summary description of the radiological environmental monitoring program;

5. At least two legible maps[2] covering all sample locations keyed to a table giving distances and directions from the centerline of one reactor;

6. The results of license participation in the Interlaboratory Comparison Program, required by Specification 3.15.3 ; and

7. Discussion of all analyses in which the LLD required by Table 4.15.1-1 was not achievable.

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

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

MONTHLY OPERATING REPORT

6.9.1.5 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office, to arrive no later than the tenth of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.6 The REPORTABLE OCCURRENCES of Specifications 6.9.1.7 and 6.9.1.8 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

6.9.1.7 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 two weeks. 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.

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 system(s) used 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.

PROMPT NOTIFICATION WITH WRITTEN FOLLOW-UP (Continued)

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 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 condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents that exceed the limits of Specification 3,11,2 or 3,12,2.

k. Exceeding the limits in Specification 3,11,5 for the storage of radioactive liquids 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.

THIRTY DAY WRITTEN REPORTS

6.9.1.8 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty 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.

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 6.9.1.7 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 gases in gaseous effluents, or 3) more than 0.05 curie of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved;
2. Cause(s) for unplanned release;
3. Actions taken to prevent recurrence; and
4. Consequences of the unplanned release.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[1]

6.9.1.9 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.1.10 These shall include the following:

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

2. A quarterly summary of hourly meteorological data collected during the reporting period in the form of joint frequency distributions of wind speed, wind direction and atmospheric stability;[2]

3. An assessment of the radiation doses to individuals due to radioactive liquid and gaseous effluents released from the station during each calendar quarter;

4. The following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

    a. Container volume
    b. Total curie quantity (specify whether determined by measurement or estimate)
    c. Principal radionuclides (specify whether determined by measurement or estimate)
    d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms)
    e. Type of container (e.g., LSA, Type A, Type B, Large Quantity)
    f. Solidification agent or absorbent (e.g., cement, urea formaldehyde)

5. A list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period; and,

6. Any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.15.2.

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

2/ In lieu of submission with the radioactive effluent release report, the licensee has the option of retaining this summary of required meteorological data in a file that shall be provided to the NRC upon request.

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 Limit of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range	Location with Highest Annual Mean		Control Locations Mean (f)[b] Range	Number of Nonroutine Reported Measurements
				Name Distance and Direction	Mean (f)[b] Range		

a. Nominal Lower Limit of Detection (LLD) as defined in

b. Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses.

SPECIAL REPORTS

Special reports may be required covering inspections, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.

6.9.2 Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director,Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within the time period specified for each report. These reports shall include the following information:

1. The cause(s) for exceeding the limit(s);
2. The action(s) taken to restore the release of radioactive effluents to be within the limit(s); and
3. A summary description of action(s) taken to prevent a similiar recurrence in the future

6.10 RECORD RETENTION

Facility records shall be retained in accordance with ANSI-N45.2.9-1974.

6.10.1 The following records shall be retained for at least five 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 OCCURRENCE submitted to the Commission.

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

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

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

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

RECORD RETENTION (Continued)

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 or 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 identified in Table 5.7.1-1.

g. Records of reactor tests and experiments.

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

i. Records of in-service inspections performed pursuant to these Technical Specifications.

j. Records of Quality Assurance activities required by the QA Manual.

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

l. Records of meetings of the PNSC and of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC)

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

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 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work 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 the facility Health Physicist in the Radiatiion Work Permit.

6.12.2 The requirements of 6.12.1, above, shall 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 Foreman on duty and/or the Plant Health Physicist.

*Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

6.13 ENVIRONMENTAL QUALIFICATION

A. 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 IE 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 DPR-71 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible 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 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

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

   a. Sufficiently detailed information to totally support rationale without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);
   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and,
   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

2. Shall become effective upon review and acceptance by the PNSC.

6.15 PROCESS CONTROL PROGRAM (PCP)

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

6.15.2 Licensee initiated changes to the PCP:

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

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;
   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and,
   c. Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

2. Shall become effective upon review and acceptance by the PNSC.

6.16 MAJOR CHANGES TO LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS[1]

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

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. 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 Part 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 that shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and admendments thereto;
   e. An evaluation of the change that shows the expected maximum exposure to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and admendments thereto;
   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;
   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 to the PNSC.

2. Shall become effective upon review and acceptance by the PNSC.

1/ Licensees may chose to submit the information called for in this Specification as part of the annual FSAR update.