ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 1
RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS
(RETS)

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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.

ACTION

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

AVERAGE PLANAR EXPOSURE

The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

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 simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CHANNEL FUNCTIONAL TEST (Continued)

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

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls in the reactor core with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative location.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be ratio of that power in the assembly which is calculated, by application of the GEXL correlation, to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μCi/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The following is defined equivalent to 1 μCi of I-131 as determined from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites": I-132, 28 μCi; I-133, 3.7 μCi; I-134, 59 μCi; I-135, 12 μCi.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 15 minutes making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

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.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable.

LIMITING CONTROL ROD PATTERN

A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually one foot. It is the integral of the heat flux over the heat transfer area associated with the unit length, usually measured in kW/ft.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor output to activated device, to ensure that components are OPERABLE.

MAXIMUM TOTAL PEAKING FACTOR

The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

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.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and are 1) described in Section 13 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

    1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

    2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.1, or

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings is) is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

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 total reactor core heat transfer rate to the reactor coolant of 2436 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE OCCURRENCE

A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.13 and 6.9.1.14

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All automatic reactor building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position.

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the reactor building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuming that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon-free.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor 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.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for two diagonal fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for two diagonal fuel bundles around each of the four SRMs.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TOTAL PEAKING FACTOR

The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

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. 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.
P	Prior to each release.
NA	Not applicable.

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 the vessel.***

**See Special Test Exception 3.10.3.

***See Special Test Exception 3.10.1.

TABLE 3.3.5.7-1 (Continued)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE FLAME	HEAT	SMOKE
3. Diesel Generator Building (Cont'd)			
Zone 7 23'	0	0	5
Zone 8 23'	0	0	5
Zone 9 23'	0	0	8
Zone 10 50'	0	0	9
4. Service Water Building			
Zone 1 4'	0	0	7
Zone 2 20'	0	0	6
5. AOG Building			
Zone 1 20'	0	0	2
Zone 2 20'	0	0	2
Zone 3 20'	1	5	1
Zone 4 37'-49'	1	6	0

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.5.8-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.8-1.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint 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 inoperable, 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 3.3.5.8-1. 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 6.9.1.14.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.5.8-1.

NOTE: See Bases 3/4.3.5.8.

TABLE 3.3.5.8-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT[a]	APPLICABILITY	ACTION
1.	Liquid Radwaste Radioactivity Effluent Monitor (Providing alarm and automatic termination of release)	*	110
2.	Liquid Radwaste Effluent Flow Measurement Device	*	111
3.	Main Service Water Effluent Radioactivity Monitor	*	112
4.	Stabilization Pond Effluent Composite Sampler	**	113
5.	Stabilization Pond Effluent Flow Measurement Device	**	114
6.	Condensate Storage Tank Level Indicating Device	*	115
7.	Service Water Effluent from Augmented Off-Gas Precooler Radioactivity Monitor	***	112
8.	Reactor Building Component Cooling Water (Service Water) Radioactivity Monitors		
	a. Effluent from Residual Heat Removal Heat Exchanger A	****	112
	b. Effluent from Residual Heat Removal Heat Exchanger B	****	112
	c. Effluent from Reactor Building Closed Cooling Water Heat Exchangers	****	112
	d. Effluent from Division I Residual Heat Removal Pump Seal Coolers	****	112
	e. Effluent from Division II Residual Heat Removal Pump Seal Coolers	****	112

TABLE 3.3.5.8-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

ACTIONS

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 4.11.1.1.2, 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 4.11.1-1. Otherwise, suspend releases via this pathway.

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 3.3.5.8-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

NOTES

* At all times

** At all times other than when the line is valved out and locked. [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].

**** At all times once these monitors are installed and become fully operational. [NOTE: These monitors are to be installed pending completion of future plant modifications.]

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

TABLE 4.3.5.8-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT[a]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1.	Liquid Radwaste Radioactivity Effluent Monitor (Providing alarm and automatic termination of release)	D	M	R[b]	Q[c]
2.	Liquid Radwaste Effluent Flow Measurement Device	D[e]	NA	R	Q
3.	Main Service Water Effluent Radioactivity Monitor	D	M	R[b]	Q[d]
4.	Stabilization Pond Effluent Composite Sampler	D	NA	R	Q
5.	Stabilization Pond Effluent Flow Measurement Device	D	NA	R	Q
6.	Condensate Storage Tank Level Indicating Device	D[f]	NA	R	Q
7.	Service Water Effluent from Augmented Off-Gas Precooler Radioactivity Monitor	D	M	R[b]	Q
8.	Reactor Building Component Cooling Water (Service Water) Radioactivity Monitors				
	a. Effluent from Residual Heat Removal Heat Exchanger A	D	M	R[b]	Q[d]
	b. Effluent from Residual Heat Removal Heat Exchanger B	D	M	R[b]	Q[d]
	c. Effluent from Reactor Building Closed Cooling Water Heat Exchangers	D	M	R[b]	Q[d]
	d. Effluent from Division I Residual Heat Removal Pump Seal Coolers	D	M	R[b]	Q[d]
	e. Effluent from Division II Residual Heat Removal Pump Seal Coolers	D	M	R[b]	Q[d]

TABLE 4.3.5.8-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

(b) 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.

(c) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument controls not set in "operate" mode.

(d) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument controls not set in "operate" mode.

(e) 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.

(f) During liquid additions to the tank.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.5.9-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology as described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.9-1.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, 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 monitoring instrumentation channel OPERABLE, take the ACTION shown in Table 3.3.5.9-1. 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 6.9.1.14.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.5.9-1.

NOTE: See Bases 3/4.3.5.8.

TABLE 3.3.5.9-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT[(a)]	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
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM		
a. Noble Gas Activity Monitor	*	123
b. Iodine Sampler Cartridge	*	127
c. Particulate Sampler Filter	*	127

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT[(a)]	APPLICABILITY	ACTION
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM (Continued)		
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 and automatic isolation)	**	121
5. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM MONITOR (Downstream of AOG Treatment System)		
a. Noble Gas Activity Monitor (providing alarm)	***	123
6. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM		
a. Recombiner Train A		
1. 1st Hydrogen Monitor	****	125
2. 2nd Hydrogen Monitor	****	125
b. Recombiner Train A		
1. 1st Hydrogen Monitor	****	125
2. 2nd Hydrogen Monitor	****	125
7. HOT SHOP VENTILATION MONITORING SYSTEM		
a. Iodine Sampler Cartridge	*	127
b. Particulate Sampler Filter	*	127

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

ACTIONS

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

a. The GASEOUS RADWASTE TREATMENT SYSTEM is not bypassed [Prior to the Augmented Off-Gas Treatment System becoming operational, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute offgas holdup line including stack filtration], 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 12 hours and these samples are analyzed for gross noble gas activity within 24 hours.

ACTION 125 - With less than two channels OPERABLE in the operating recombiner train, operation of the train may continue provided proper function of the recombiner is assured by monitoring recombiner temperature in accordance with approved procedures.

With less than one channel OPERABLE in the operating recombiner train, operation of the train may continue provided grab samples from the 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.

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 4.11.2-1.

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

NOTES

* At all times.

** During operation of the main condenser air ejector.

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

**** At all times during recombiner train operation.

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

TABLE 4.3.5.9-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT[a]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	MAIN STACK MONITORING SYSTEM					
	a. Noble Gas Activity Monitor	D	M	R[b]	Q[d]	*
	b. Iodine Sampler Cartridge	W	NA	NA	NA	*
	c. Particulate Sampler Filter	W	NA	NA	NA	*
	d. System Effluent Flow Rate Measurement Device	D	NA	R	Q	*
	e. Sampler Flow Rate Measurement Device	D	NA	R	Q	*
2.	REACTOR BUILDING VENTILATION MONITORING SYSTEM					
	a. Noble Gas Activity Monitor	D	M	R[b]	Q[d]	*
	b. Iodine Sampler Cartridge	W	NA	NA	NA	*
	c. Particulate Sampler Filter	W	NA	NA	NA	*
	d. System Effluent Flow Rate Measurement Device	D	NA	R	Q	*
	e. Sampler Flow Rate Measurement Device	D	NA	R	Q	*

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT[a]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R[b]	Q[d]	*
b. Iodine Sampler Cartridge	W	NA	NA	NA	*
c. Particulate Sampler Filter	W	NA	NA	NA	*
d. System Effluent Flow Rate Measurement Device	D	NA	R	Q	*
e. Sampler Flow Rate Measurement Device	D	NA	R	Q	*
4. MAIN CONDENSER AIR EJECTOR RADIOACTIVITY MONITOR (Prior to input to treatment system)					
a. Noble Gas Activity Monitor (Providing alarm and automatic isolation)	D	M	R[b]	Q[c]	**
5. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM MONITOR (Downstream of AOG Treatment System)[f]					
a. Noble Gas Activity Monitor (Providing alarm)	D	P	R[b]	Q	***

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT[(a)]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM[(f)]					
a. Recombiner Train A					
1. 1st Hydrogen Monitor	D	NA	Q[(e)]	M	****
2. 2nd Hydrogen Monitor	D	NA	Q[(e)]	M	****
b. Recombiner Train B					
1. 1st Hydrogen Monitor	D	NA	Q[(e)]	M	****
2. 2nd Hydrogen Monitor	D	NA	Q[(e)]	M	****
7. HOT SHOP VENTILATION MONITORING SYSTEM					
a. Iodine Sampler Cartridge	W	NA	NA	NA	*
b. Particulate Sampler Filter	W	NA	NA	NA	*

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

(b) 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.

(c) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument not set in "operate" mode.

(d) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument not set in "operate" mode.

(e) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

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

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

* At all times other than when the line is valved out and locked.

** During operation of the main condenser air ejector.

*** At all times other than when the line is valved out and locked (once the Augmented Off-Gas Treatment System becomes fully operational).

**** During recombiner train operation.

INSTRUMENTATION

3/4.3.6 ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation trip systems shown in Table 3.3.6.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.1-2.

APPLICABILITY: CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6.1-2, declare the trip system inoperable until the trip system is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the minimum number of OPERABLE trip systems per operating pump not satisfied for one Trip Function, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.6.1.1 Each ATWS recirculation pump trip system instrumentation trip system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.1.1-1.

4.3.6.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

TABLE 3.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE TRIP SYSTEMS PER OPERATING PUMP
1. Reactor Vessel Water Level - Low, Level 2 (B21-LT-N024A-2,B-2 and B21-LT-N025A-2,B-2) (B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	1
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	1

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 2 (B21-LTM-N024A-2,B-2; B21-LTM-N025A-2,B-2)	$\geq$ +112 inches*	$\geq$ +112 inches*
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	$\leq$ 1120 psig	$\leq$ 1120 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 4.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level-Low, Level 2			
(B21-LT-N024A-2,B-2 and B21-LT-N025A-2,B-2)	NA[(a)]	NA	R[(b)]
(B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	D	M	M
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	NA	M	R

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In CONDITION 1, 2, and 3:

   1. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1, but less than 10 μmho/cm at 25°C and less than 0.5 ppm, respectively, operation may continue for up to 24 hours and this condition need not be reported to the Commission per Specification 6.9.1.12 provided that operation under these conditions shall not exceed 336 hours per year. The provisions of Specification 3.0.4 are not applicable.

   2. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1 for more than 24 hours during one continuous time interval or with the conductivity exceeding 10 μmho/cm at 25°C or chloride exceeding 0.5 ppm, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. At all other times with the conductivity and/or chloride concentration of the reactor coolant in excess of the limit specified in Table 3.4.4-1, restore the conductivity and/or chloride concentration to within the limit within 48 hours.

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the reactor coolant shall be limited to:

a. $\leq$ 0.2 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq 100/\bar{E}$ μCi/gram.

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

ACTION:

a. In CONDITION 1, 2, and 3, with the specific activity of the reactor coolant;

1. > 0.2 μCi/gram DOSE EQUIVALENT I-131 but $\leq$ 4.0 μCi/gram, operation may continue for up to 48 hours provided that operation under these conditions shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

2. > 0.2 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or > 4.0 μCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. > $100/\bar{E}$ μCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In CONDITION 1, 2, 3, or 4,

1. With the specific activity of the primary coolant > 0.2 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 at least once per 4 hours until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE report shall be prepared and submitted to the Commission pursuant to Specification 6.9.1.12. This report shall contain the results of the specific activity analyses and the time duration when the specific activity coolant exceeded 0.2 μCi/gram DOSE EQUIVALENT I-131 together with the below additional information.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The structural integrity of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection to the accessible interior and exterior surfaces of the containment and verifying no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the primary containment detected during the required inspections shall be reported to the Commission pursuant to Specification 6.9.1.12.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (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

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1-1. If the stabilization pond or service water samples analyzed according to Table 4.11.1-1 indicate concentrations of any gamma-emitting radionuclides greater than 5×10^{-6} µCi/ml (trigger level), then the liquid wastes exceeding the trigger level shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1-2 until such time as the sample concentration of each gamma-emitting nuclide is less than 5×10^{-6} µCi/ml.

4.11.1.1.2 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 3.11.1.1.

NOTE: See Bases 3/4.11.1.1

TABLE 4.11.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml) (a)(e)
A.1. 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)
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (gamma emitters)	1×10^{-5}
2. Circulating Water Pit	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(d)	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(d) (Potential Continuous Release)	W During System Operation	W During System Operation	Principal Gamma Emitters(g)	5×10^{-7} (b)

TABLE 4.11.1-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

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 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) The stabilization pond and service water liquid release types represent potential release pathways and not actual release pathways. Surveillance of these pathways is intended to alert the plant to a potential problem; analysis for principal gamma emitters should be sufficient to meet this intent. If analysis for principal gamma emitters indicates a problem (i.e., exceeds the trigger level of 5×10^{-6} µCi/ml), then complete sampling and analyses shall be performed as per Table 4.11.1-2.

(e) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$\text{LLD} = \frac{4.66\ \sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

TABLE 4.11.1-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

N = background count rate (cpm)

t_b = time background counted for (min)

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

(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 composite sampling instrumentation becomes available and is OPERABLE, daily grab sampling of the stabilization pond effluent will not be required during release and the composite sample will be analyzed on a weekly basis.

(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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

(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 4.11.1-1.

TABLE 4.11.1-2

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
FOR POTENTIAL RELEASE PATHWAYS WHICH HAVE EXCEEDED TRIGGER LEVELS

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

TABLE 4.11.1-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
FOR POTENTIAL RELEASE PATHWAYS WHICH HAVE EXCEEDED TRIGGER LEVELS

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 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) Until such time as continuous proportional composite samplers are installed on the service water discharge line, daily grab sampling of the service water effluent will be required for use in making up the composite.

(e) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

Where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

N = background count rate (cpm)

t_b = time background counted for (min)

TABLE 4.11.1-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
FOR POTENTIAL RELEASE PATHWAYS WHICH HAVE EXCEEDED TRIGGER LEVELS

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

(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 composite sampling instrumentation becomes available and is OPERABLE, daily grab sampling of the stabilization pond effluent will not be required during release and the composite sample will be analyzed on a weekly basis.

(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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

(h) A continuous release is the discharge of liquid waste of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

DOSE - LIQUID EFFLUENTS

LIMITING CONDITION FOR OPERATION

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

a. During any calendar quarter to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ, and

b. During any calendar year to less than or equal to 6 mrem to the total body and to less than or equal to 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 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(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

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

NOTE: See Bases 3/4.11.1.2

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 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 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 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases from the site to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the ODCM.

NOTE: See Bases 3/4.11.1.3

RADIOACTIVE EFFLUENTS

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

3.11.1.4 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	10 Ci
b. 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 above limit, without delay suspend all addition of radioactive material to the tank, within 48 hours reduce the tank's contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the tanks listed shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

NOTE: See Bases 3.4.11.1.4

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site 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, tritium, and 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).

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 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 4.11.2-1.

NOTE: See Bases 3/4.11.2.1

TABLE 4.11.2-1

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 Emmitters[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[h]	M[c][d] Grab Sample	M[c]	Principal Gamma Emmitters[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 Emmitters[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 4.11.2-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATION

(a) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$\text{LLD} = \frac{4.66\sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

Where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

N = background count rate (cpm)

t_b = time background counted for (min)

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

Typical values of E. V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

(b) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for

particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

(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 spent 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. Spent 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 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(f) Sample 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.

(h) Monthly grab samples to be analyzed for principal gamma emitters and tritium are not applicable for the Hot Shop environmental release point. In addition, the Hot Shop release point does not have a continuous noble gas monitor and, therefore, the noble gas activity analysis requirements of Table 4.11.2-1 are not applicable.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 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: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation;

b. During any calendar year: Less than or equal to 20 mrad for gamma radiation and less than or equal to 40 mrad for beta radiation.

APPLICABILITY: At all times.

ACTIONS:

a. With the calculated air dose from radioactive noble gases 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(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

4.11.2.2 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.

NOTE: See Bases 3/4.11.2.2

DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 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

4.11.2.3 Dose Calculations - Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-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.

NOTE: See Bases 3/4.11.2.3

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector (evacuation) system is in operation.

ACTION:

a. With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, 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. Identification of the inoperable equipment or subsystems and the reason for 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

4.11.2.4 The readings of the relevant instruments shall be checked at least once per 12 hours when the main condenser air ejector is in use to ensure that the GASEOUS RADWASTE TREATMENT SYSTEM is functioning.

NOTE: See Bases 3/4.11.2.4

RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.5 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 other than when the VENTILATION EXHAUST TREATMENT SYSTEM is undergoing routine maintenance.

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. 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

4.11.2.5 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM, when the VENTILATION EXHAUST TREATMENT SYSTEM is not in use.

NOTE: See Bases 3/4.11.2.5

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.*

ACTION:

a. With the concentration of hydrogen in the main condenser offgas 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

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limit by continuously monitoring the waste gases in the main condenser offgas treatment system with the hydrogen monitors required OPERABLE by Table 3.3.5.9-1 of Specification 3.3.5.9.

NOTE: See Bases 3/4.11.2.6

*This specification shall become applicable when the offgas recombiners become operational.

MAIN CONDENSER AIR EJECTOR RADIOACTIVITY RELEASE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.7 The release rate of the sum of the activities from the noble gases measured at the main condenser air ejector shall be limited to less than equal to 243,600 microcuries/second (the Kr-85m, 87, 88 and Xe-133, 135, 138 contribution after 30 minutes decay).

APPLICABILITY: During operation of the main condenser air ejector.

ACTION:

With the release rate of the sum of the activities from the 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

4.11.2.7.1 The radioactivity rate of noble gases at (near) the outlet of the main condenser air ejector shall be continuously monitored in accordance with Specification 3.11.2.1.

4.11.2.7.2 The release rate of the sum of the activities from the 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, or

b. Within 31 days following each refueling/maintenance outage, and

c. Within 4 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, of 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.

NOTE: See Bases 3/4.11.2.7

DRYWELL VENTING OR PURGING

LIMITING CONDITION FOR OPERATION

3.11.2.8 The drywell shall be purged to the environment at a rate in conformance with Specification 3.11.2.1.

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

4.11.2.8 A sample analysis, as defined in Table 4.11.2-1, shall be performed prior to each drywell PURGE.

NOTE: See Bases 3/4.11.2.8

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 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 6.9.1.14.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 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 6.14, to assure SOLIDIFICATION of subsequent batches of waste.

NOTE: See Bases 3/4.11.3

3/4.11.4 TOTAL DOSE (40 CFR PART 190)

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and 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 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made which, in addition to doses due to effluents, include direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, 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.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE (40 CFR PART 190)

SURVEILLANCE REQUREMENTS

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

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

NOTES: See Bases 3/4.11.4

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, 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 sampling medium at a specific location exceeding the reporting levels of Table 3.12.1-2 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 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 3.12.1-2 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report. 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 Operating Report.

c. With milk or fresh leafy vegetables unavailable from one or more of the sample locations required by Table 3.12.1-1, identify locations for obtaining replacement samples and add them to the radiological

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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 6.9.1.8, 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

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 3.12.1-1 and the detection capabilities required by Table 4.12.1-1.

NOTE: See Bases 3/4.12.1

TABLE 3.12.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[(a)]	Sampling and Collection Frequency	Type and Frequency Of Analysis
1. DIRECT RADIATION[(b)]	40 Locations. At each location with 2 or more dosimeters or one or more instruments for continuously measuring and recording dose rate, placed as follows: An inner ring of stations, with at least one in each meteorological sector in the general area of the site boundary as is reasonably accessible and practical; An outer ring of stations, with at least one in each meteorological sector at distances of 8 km or greater from the site as is reasonably accessible and practical; and The balance of stations to be placed in special interest areas such as population centers, nearby residences, schools, and in at least one or two areas to serve as control stations.	Q	Gamma Dose - Q

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[(a)]	Sampling and Collection Frequency	Type and Frequency Of Analysis
2. AIRBORNE - Radioiodine and Particulate	5 Locations, as follows: 3 samples from different sectors as close to the SITE BOUNDARY as is reasonably accessible, one of which being at the highest calculated annual average ground level D/Q; 1 sample from the vicinity of a nearby community; and 1 sample from a control location, as for example greater than 15 km distant and in a less prevalent wind direction[(c)]	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent.	Radioiodine Cannister: I-131 analysis - W Particulate sampler: Gross beta radioactivity analysis following filter change;[(d)] Gamma isotopic analysis[(e)] of composite (by location) - Q
3. WATERBORNE a. Surface[(f)]	2 locations, as follows: 1 sample upstream 1 sample downstream	Composite[(g)] sample collection - M	Gamma Isotopic Analysis[(e)] - M Tritium Analysis - Q

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[(a)]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE (Continued)			
b. Sediment from shoreline	1 location with a sample taken from a downstream area with existing or potential recreational value	SA	Gamma Isotopic Analysis[(e)] - SA
4. INGESTION			
a. Milk	4 locations as follows: Samples from milking animals at 3 locations within 8 km distance having the highest dose potential (when available)[(h)] 1 sample from milking animals at a control location greater than 15 km distance from the site and in a less prevalent wind direction	With animals on pasture - SM At other times - M	Gamma isotopic analysis[(e)] and I-131 analysis - SM (when animals are on pasture); At other times - M
b. Fish and Invertebrates	4 locations as follows: 3 samples of commercially and recreationally important species in the vicinity of the plant discharge: one free swimming	When in season - SA	Gamma isotopic analysis[(e)] on edible portions - SA

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[(a)]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION (Continued)			
b. Fish and Invertebrates (Continued)	species; one bottom feeding species; and one shellfish species. 1 sample of a similarly edible species from an area not influenced by plant discharge to serve as a control sample.		
c. Broadleaf Vegetation	3 locations as follows: Samples of broadleaf vegetation grown in 2 sectors of historically higher D/Q values at the SITE BOUNDARY if milk sampling is not performed. 1 sample of a similar broadleaf vegetation grown at a distance of greater than 15 km from the site in a less prevalent wind direction if milk sampling is not performed.	When available - M	Gamma isotopic analysis[(e)] and I-131 analysis - M (when available)

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATION

(a) Specific parameters of distance and direction sector from the site, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a License Event Report and pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples for that pathway 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).

(b) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

(c) The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

(d) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

(e) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(f) The "upstream" sample shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

(g) A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. 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 compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(h) When less than three (3) milking animals are available for testing within an 8-km distance, sampling of broadleaf vegetation shall be performed as indicated in Table 3.12.1-1, 4.c, in lieu of milk sampling.

TABLE 3.12.1-2 (Continued)

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)	Broadleaf Vegetation (pCi/kg, wet)
H-3	3×10^4	-	-	-	-
Mn-54	1×10^3	-	3×10^4	-	-
Fe-59	4×10^2	-	1×10^4	-	-
Co-58	4×10^2	-	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 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Broadleaf Vegetation (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01	-	-	-	-
H-3	3000	-	-	-	-	-
Mn-54	15	-	130	-	-	-
Fe-59	30	-	260	-	-	-
Co-58, 60	15	-	130	-	-	-
Zn-65	30	-	260	-	-	-
Zr-Nb-95	15	-	-	-	-	-
I-131	1[c]	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	-	-

TABLE 4.12.1-1 (Continued)

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

TABLE NOTATION

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

(b) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

Where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

N = background count rate (cpm)

t_b = time background counted for (min)

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

TABLE 4.12.1-1 (Continued)

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

(c) The LLD of gamma isotopic analysis may be used.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 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 3.12.1-1(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 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

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 3.12.1, 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 6.9.1.8, 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

4.12.2 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 6.9.1.7.

NOTE: See Bases 3/4.12.2

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

NOTE: See Bases 3/4.12.3

MONITORING INSTRUMENTATION (Continued)

3/4.3.5.6 CHLORINE INTRUSION MONITORS

The chloride intrusion monitors provide adquate warning of any leakage in the condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. With only a minimum number of instruments available, increased sampling frequency provides adequate information for the same purpose.

3/4.3.5.7 FIRE DETECTION INSTRUMENTATION

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

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

3/4.3.5.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the 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 limiting condition for operation is being exceeded, takes the next appropriate action to comply with the specification.

3/4.3.5.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring 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

MONITORING INSTRUMENTATION (Continued)

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, the main condenser offgas treatment system monitor, and the explosive gas monitoring instrumentation shown in Table 3.3.5.9-1 are not considered effluent monitoring instrumentation in the same sense as the other instrumentation listed in the Table. Therefore, their alarm/trip setpoints are not necessarily set to ensure that the limits of Specification 3.11.2.1 are not exceeded.

The main condenser air ejector monitoring instrumentation channels are provided to monitor and control gross radioactivity removed from the main condenser. The alarm/trip setpoints for the main condenser air ejector monitor are set to ensure that the limits of Specification 3.11.2.7 are not exceeded. 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.

This specification also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the offgas treatment system (hydrogen monitors). The hydrogen monitors will become applicable when the offgas recombiners become fully operational (prior to operation of the Augmented Off-Gas Treatment System) 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.

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

3/4.3.6 ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The ATWS recirculation pump trip system has been added at the suggestion of ACRS as a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events given in General Electric Company Topical Report NEDO-10349, dated March, 1971.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents 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 HASL 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 limiting condition for operation is being exceeded, takes the next appropriate action to comply with the specification.

Note that for batch releases, recirculation of at least two tank volumes shall be considered adequate for thorough mixing.

The stabilization pond and service water liquid release types represent potential release pathways and not actual release pathways. Surveillance of these pathways is intended to alert the plant to a potential problem; analysis for principal gamma emitters should be sufficient to meet this intent. If analysis for principal gamma emitters indicates a problem (i.e., exceeds the trigger level of 5×10^{-6} µCi/ml), then complete sampling and analyses shall be performed as per Table 4.11.1-2. The trigger level of 5×10^{-6} µCi/ml was chosen as being sufficient to provide reasonable assurance of accountability of all nuclides released based upon lower limits of detection and expected concentrations.

3/4.11.1.2 DOSE - LIQUID EFFLUENTS

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

DOSES (Continued)

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 to UNRESTRICTED AREAS 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. This specification is written for a two unit site.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

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 10 CFR 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 two reactor units having a shared liquid radwaste treatment system.

3/4.11.1.4 LIQUID HOLDUP TANKS

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 the tank and its associated piping are all Seismic Class I.

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.

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

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

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 for UNRESTRICTED AREAS. 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 in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II, of 10 CFR Part 20 [10 CFR Part 20.106 (b)]. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC 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 a MEMBER OF THE PUBLIC 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.

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).

DOSE RATE (Continued)

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

3/4.11.2.2 DOSE-NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and, at the same time, implement the guides set forth in Section IV.A of Appendix I, to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS 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. The limits of this specification are twice the 10 CFR 50 Appendix I per reactor guidelines because they are written for a two unit site.

3/4.11.2.3 DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

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 to UNRESTRICTED AREAS 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

DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

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. The limits of this specification are twice the 10 CFR 50 Appendix I per reactor guidelines because they are written for a two unit site.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

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.

Until such time as the Augmented Off-Gas Treatment System becomes operational at the Brunswick Steam Electric Plant, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute offgas holdup line and stack filter house filtration. After the Augmented Off-Gas Treatment System becomes operational, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute offgas holdup line, stack filter house filtration, and the Augmented Off-Gas Treatment System.

3/4.11.2.5 VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTILATION EXHAUST TREATMENT SYSTEM (Continued)

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. At the Brunswick Steam Electric Plant, the only VENTILATION EXHAUST TREATMENT SYSTEMS shall be those installed for the Turbine Buildings' ventilation.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas 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 Off-Gas Recombiners become fully operational (prior to operation of the Augmented Off-Gas Treatment System) 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.

3/4.11.2.7 MAIN CONDENSER AIR EJECTOR RADIOACTIVITY RELEASE RATE

Restricting the release rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at or beyond the exclusion boundary 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. 243,600 microcuries/second is equal to 100 microcuries/second/MWt for a rated thermal power of 2,436 MWt.

3/4.11.2.8 DRYWELL VENTING OR PURGING

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.

3/4.11.3 SOLID RADIOACTIVE WASTE

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.

TOTAL DOSE (40 CFR PART 190) (Continued)

3/4.11.4 TOTAL DOSE (40 CFR PART 190)

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 generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. 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 fulfills 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 Specification 3/4.11. 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.

3/4.12.1 MONITORING PROGRAM

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 effluent 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 analyses are tabulated in terms of the Lower Limits of Detection (LLDs). The LLDs required by Table 4.12.1-1 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 Determination 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.

3/4.12.2 LAND USE CENSUS

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.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

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.

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 CONTAINMENT

CONFIGURATION

5.2.1 The PRIMARY CONTAINMENT is a steel-lined reinforced concrete structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a concrete steel-lined pressure vessel in the shape of a torus. The primary containment has a minimum free air volume of (288,000) cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 62 psig.

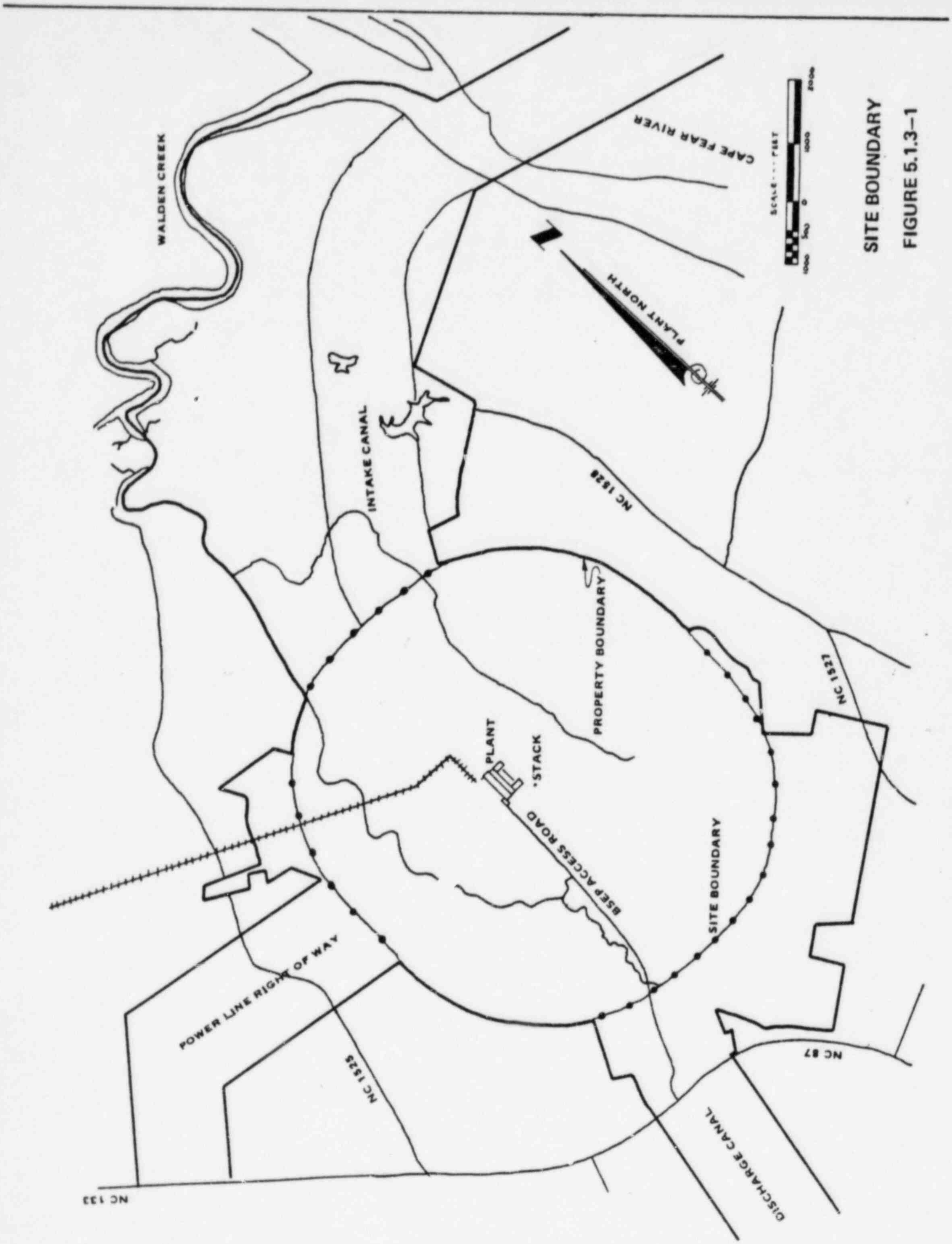
b. Maximum internal temperature: drywell 300°F.
suppression chamber 200°F.

c. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 560 fuel assemblies, with each fuel assembly containing 63 fuel rods clad with Zircaloy 2. Each fuel rod shall have a nominal active fuel length of 146 inches for 8 x 8 fuel and 150 inches for 8 x 8R fuel and contain a maximum total weight of 3,355 grams of UO_2. The initial core loading.



SITE BOUNDARY

FIGURE 5.1.3-1

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant 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. The facility duty shift shall be composed of at least the minimum facility shift crew composition shown in Table 6.2.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor.

c. When either reactor is in CONDITION 1, 2, or 3, at least one licensed Senior Reactor Operator shall be in the control room.

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.

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* The individual qualified to implement radiation protection procedures and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

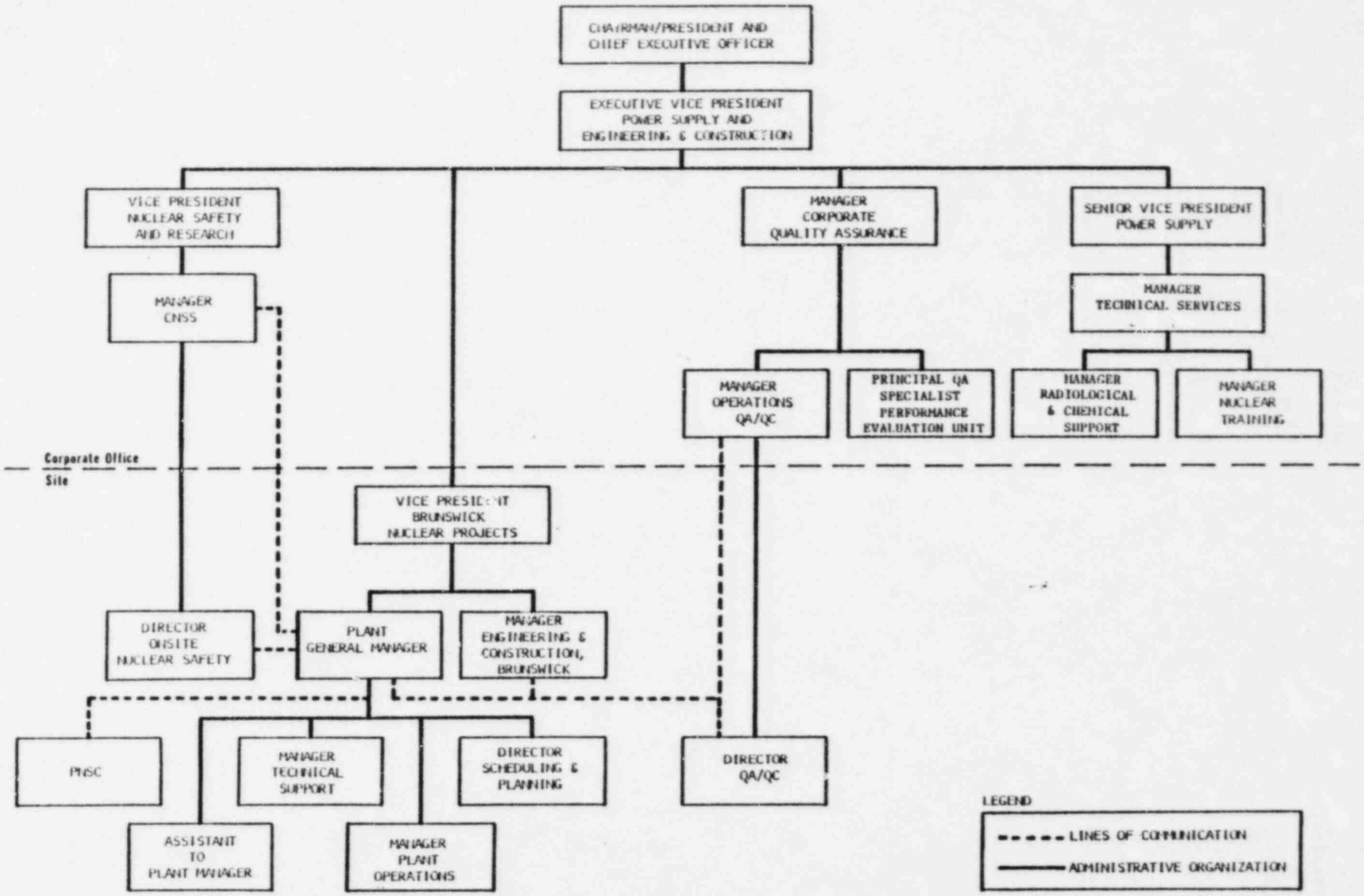
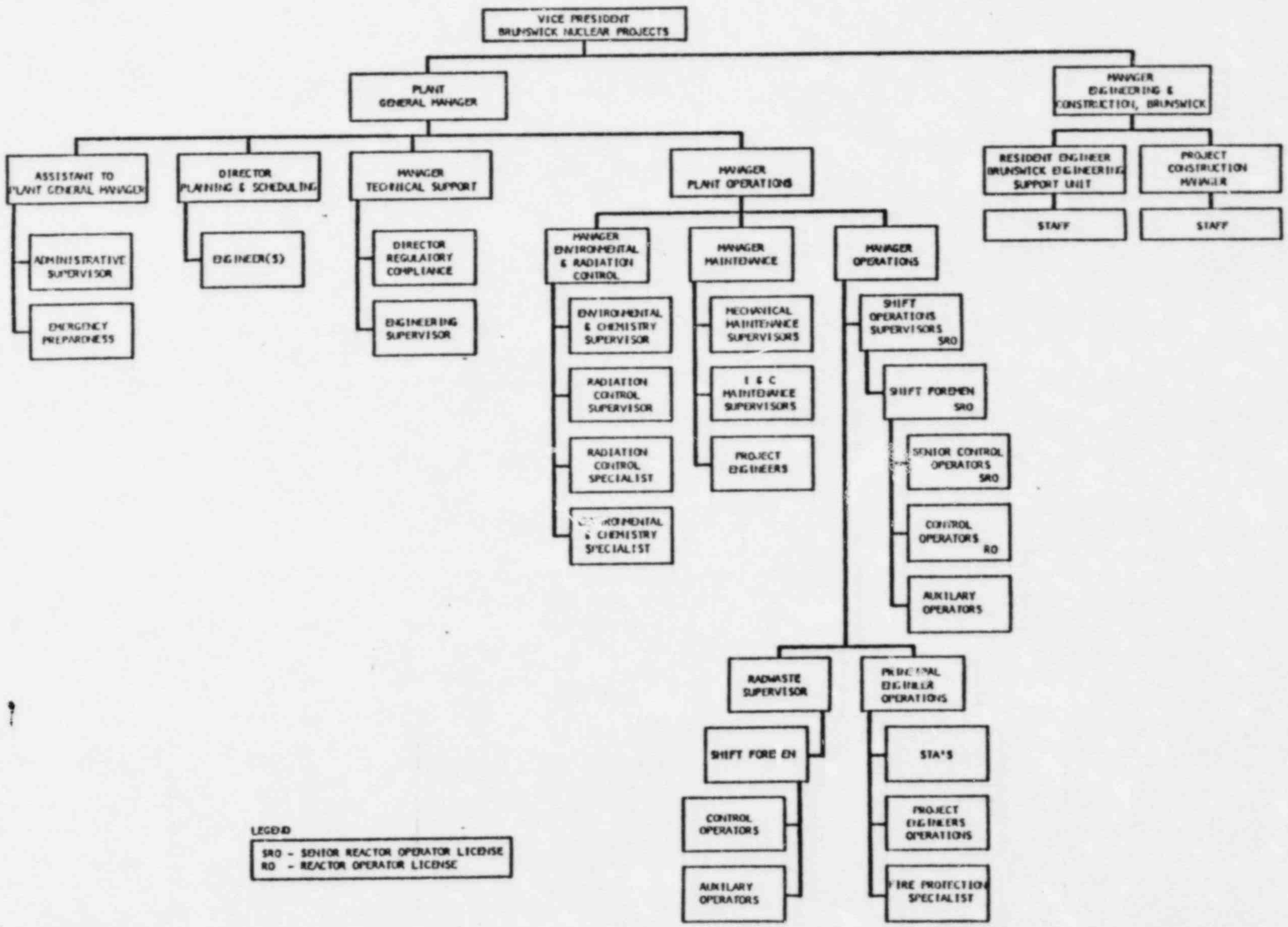

CHAIRMAN/PRESIDENT AND CHIEF EXECUTIVE OFFICER
EXECUTIVE VICE PRESIDENT POWER SUPPLY AND ENGINEERING & CONSTRUCTION
VICE PRESIDENT NUCLEAR SAFETY AND RESEARCH
MANAGER CORPORATE QUALITY ASSURANCE
SENIOR VICE PRESIDENT POWER SUPPLY
MANAGER CNSS
MANAGER TECHNICAL SERVICES
MANAGER OPERATIONS QA/QC
PRINCIPAL QA SPECIALIST PERFORMANCE EVALUATION UNIT
MANAGER RADIOLOGICAL & CHEMICAL SUPPORT
MANAGER NUCLEAR TRAINING
Corporate Office
Site
VICE PRESIDENT BRUNSWICK NUCLEAR PROJECTS
DIRECTOR ONSITE NUCLEAR SAFETY
PLANT GENERAL MANAGER
MANAGER ENGINEERING & CONSTRUCTION, BRUNSWICK
PNSC
MANAGER TECHNICAL SUPPORT
DIRECTOR SCHEDULING & PLANNING
DIRECTOR QA/QC
ASSISTANT TO PLANT MANAGER
MANAGER PLANT OPERATIONS
LEGEND
LINES OF COMMUNICATION
ADMINISTRATIVE ORGANIZATION

Figure 6.2.2-1

FACILITY ORGANIZATION



PLANT FIRE PROTECTION ORGANIZATION
BRUNSWICK STEAM ELECTRIC PLANT

Figure 6.2.2-2

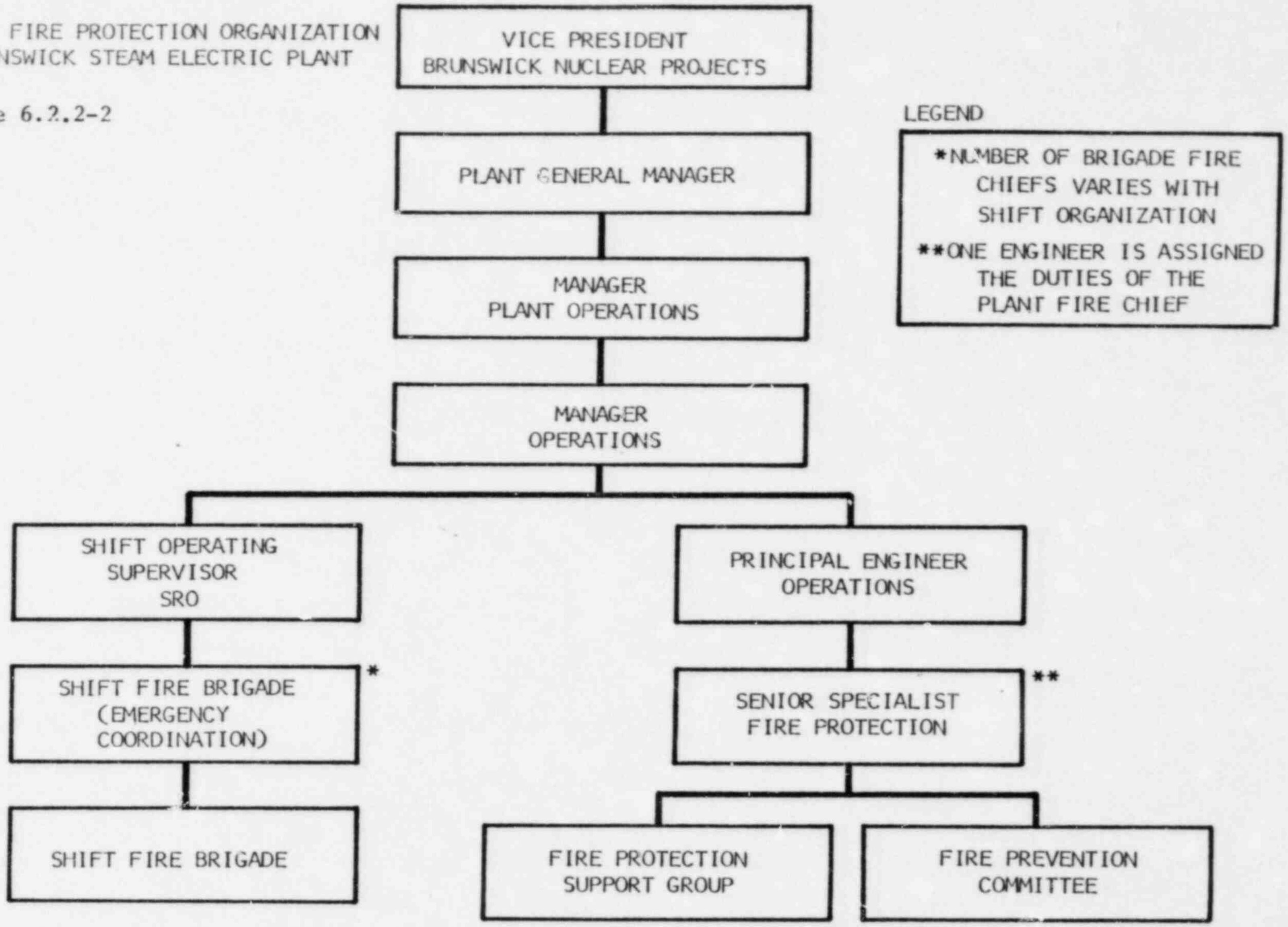


TABLE 6.2.2-1
MINIMUM FACILITY SHIFT CREW COMPOSITION

WITH UNIT 2 IN CONDITION 1, 2, OR 3		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[b]
SRO[a]	1	1[b]
RO[a]	3	2
AO	3	3
STA	1	1

WITH UNIT 2 IN CONDITION 4 OR 5		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[b]
SRO[a]	1	1[b]
RO[a]	3	3
AO	3	3
STA	1	None

WITH UNIT 2 DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[b]
SRO[a]	1	1[b]
RO[a]	2	2
AO	3	3
STA	1	None

TABLE 6.2.2-1 (Continued)

MINIMUM FACILITY SHIFT CREW COMPOSITION

TABLE NOTATION

SOS - Shift Operating Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Auxiliary Operator (non-licensed individual)
STA - Shift Technical Advisor

(a) Assumes each individual is licensed on both plants.

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

The Shift Crew Composition 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.

6.2.3 ONSITE NUCLEAR SAFETY (ONS)

FUNCTION

6.2.3.1 The ONS Unit shall function to examine facility operating characterisitics, NRC issues, industry advisories, and other sources which may indicate areas for improving facility safety.

RESPONSIBILITIES

6.2.3.2 The ONS Unit shall be responsible for maintaining surveillance of facility activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.3 The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving facility safety to the Manager-Corporate Nuclear Safety.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Operating Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.3 FACILITY STAFF QUALIFICATION

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Manager - Environmental & Radiation Control who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

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* Not responsible for sign-off function.

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 Manager-Operations and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 NUCLEAR SAFETY REVIEWERS

6.5.1.1 Individuals shall be designated/approved by the Plant General Manager for the first party and second party safety reviews.

6.5.1.2 Individuals designated under 6.5.1.1 above shall have an academic degree in an engineering or related field or equivalent and two years related experience.

6.5.1.3 A list shall be maintained of individuals qualified to perform first party and second party safety reviews.

6.5.1.4 The list specified in 6.5.1.3 above shall include individuals, in addition to the first party and second party safety reviewers, whose expertise may be necessary during the reviews to assure that the reviewers collectively possess the background and qualfications in the disciplines necessary and important to the specific review.

6.5.1.5 The list specified in 6.5.1.3 and 6.5.1.4 above shall include the disciplines for which each individual is qualified.

6.5.2 SAFETY REVIEW AND CONTROL

PROCEDURES, TESTS, AND EXPERIMENTS

6.5.2.1 A safety evaluation for the procedures required by Specification 6.8, other procedures that affect nuclear safety, and changes thereto shall be prepared.

6.5.2.2 A safety evaluation for proposed tests and experiments that affect nuclear safety shall be prepared.

PROCEDURES, TESTS, AND EXPERIMENTS (Continued)

6.5.2.3 The safety evaluation prepared in accordance with 6.5.2.1 or 6.5.2.2 above shall include a written determination, with basis, of whether or not the procedures required by Specification 6.8, other procedures that affect nuclear safety, proposed tests and experiments, and changes thereto constitutes an unreviewed safety question as defined in 10 CFR 50.59, or involves a change to the Technical Specifications.

6.5.2.4 A first party safety review of the safety evaluation prepared in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 above shall be performed by an individual qualified under Section 6.5.1. The first party reviewer may be the preparer of the safety evaluation.

6.5.2.5 A second party safety review of the safety evaluation prepared and reviewed in accordance with 6.5.2.1, 6.5.2.2, 6.5.2.3, and 6.5.2.4 above shall be performed prior to approval and implementation of the proposed changes.

6.5.2.6 The second party safety review shall be performed by an individual qualified under Section 6.5.1. The second party safety reviewer shall be an individual other than the individual who was the original preparer or the first party safety reviewer.

6.5.2.7 Following the first party and second party safety review, the procedures required by Specification 6.8, other procedures that affect nuclear safety, proposed tests or experiments, and changes thereto (other than editorial or typographical) which have been determined to not involve an unreviewed safety question as defined in 10 CFR 50.59 or change to the Technical Specifications shall be approved prior to implementation by the Plant General Manager, the designated alternate to the Plant General Manager, or the Manager of the functional area affected by the procedure, proposed test or experiment. These procedures, proposed tests or experiments, and changes thereto shall be approved by an individual other than those who performed the required safety reviews.

6.5.2.8 The procedures required by Specification 6.8 and changes thereto, other procedures that affect nuclear safety and changes thereto, and proposed tests or experiments and changes thereto that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications shall be reviewed by the Plant Nuclear Safety Committee and submitted to the Commission for approval prior to implementation. Matters of this kind shall be referred to the Corporate Nuclear Safety Section by the Plant General Manager or by other functional organizational units within Carolina Power & Light Company for review prior to implementation.

6.5.2.9 The procedures required by Specification 6.8 and changes thereto, any other procedures that affect nuclear safety and changes thereto, and proposed tests or experiments that constitute a change to the safety analysis report shall be referred to the Corporate Nuclear Safety Section for review pursuant to Specification 6.5.4.9.

PROCEDURES, TESTS, AND EXPERIMENTS (Continued)

6.5.2.10 Temporary changes to procedures required by Specification 6.8, any other procedures that affect nuclear safety, and proposed tests or experiments may be approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the unit affected, if such a change does not change the intent of the original procedure, proposed test or experiment. Temporary changes shall be documented and, within 21 days of receiving approval, be reviewed pursuant to 6.5.2.1, 6.5.2.2, 6.5.2.3, 6.5.2.4, 6.5.2.5, 6.5.2.6, and 6.5.2.7 above.

MODIFICATIONS

6.5.2.11 A safety evaluation for all proposed modifications to plant systems or equipment that affect nuclear safety shall be performed.

6.5.2.12 The safety evaluation prepared shall include a written determination, with basis, of whether or not the proposed modification is a change in the facility as described in the safety analysis report, involves a change to the Technical Specifications, or constitutes an unreviewed safety question as defined in 10 CFR 50.59.

6.5.2.13 A first party safety review of the safety evaluation prepared in accordance with 6.5.2.11 and 6.5.2.12 above shall be performed by an individual qualified under Section 6.5.1. The first party safety reviewer may be the preparer of the safety evaluation.

6.5.2.14 A second party safety review of the safety evaluation prepared and reviewed in accordance with 6.5.2.11, 6.5.2.12, and 6.5.2.13 above shall be performed prior to approval of the proposed modification.

6.5.2.15 The second party safety review shall be performed by an individual qualified under Section 6.5.1. The second party safety reviewer shall be an individual other than the individual who was the original preparer or the first party safety reviewer.

6.5.2.16 Following the first party and second party safety review, proposed modifications which have been determined to not involve an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications shall be approved prior to implementation by the Plant General Manager or the designated alternate of the Plant General Manager. The proposed modifications shall be approved by an individual other than those who performed the required safety reviews.

6.5.2.17 Proposed modifications that are determined to constitute an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications shall be reviewed by the Plant Nuclear Safety Committee and submitted to the Commission for approval prior to implementation. Matters of this kind shall be referred to the Corporate Nuclear Safety Section for review prior to implementation.

MODIFICATIONS (Continued)

6.5.2.18 Proposed modifications that constitute a change to the safety analysis report shall be referred to the Corporate Nuclear Safety Section for review pursuant to Specification 6.5.4.9.

OPERATING LICENSE CHANGES

6.5.2.19 A safety evaluation for all proposed changes to the Technical Specifications and the Operating License shall be prepared.

6.5.2.20 The safety evaluation prepared shall include a written preliminary determination, with basis, of whether or not the proposed Technical Specification/Operating License change(s) is a change in the facility as described in the safety analysis report.

6.5.2.21 A first party safety review of the safety evaluation prepared in accordance with 6.5.2.19 and 6.5.2.20 above shall be performed by an individual qualified under Section 6.5.1. The first party safety reviewer may be the preparer of the safety evaluation.

6.5.2.22 A second party safety review of the safety evaluation prepared and reviewed in accordance with 6.5.2.19, 6.5.2.20, and 6.5.2.21 above shall be performed prior to approval of the proposed change(s) by the Plant Nuclear Safety Committee.

6.5.2.23 The second party safety review shall be performed by an individual qualfied under Section 6.5.1. The second party safety reviewer shall be an individual other than the individual who was the original preparer or the first party safety reviewer.

6.5.3 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.3.1 As an effective means for the regular review, overview, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) shall be established.

6.5.3.2 The PNSC shall function through the utilization of subcommittees, audits, investigations, reports, and/or performance of reviews as a group.

COMPOSITION

6.5.3.3 The PNSC shall be composed of the:

Chairman:	Plant General Manager*
Member:	Manager - Plant Operations
Member:	Manager - Technical Support
Member:	Manager - Operations
Member:	Manager - Maintenance
Member:	Manager - Environmental & Radiation Control
Member:	Assistant to Plant General Manager
Member:	Director - QA/QC
Member:	Director - Regulatory Compliance

ALTERNATES

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

6.5.3.5 All alternates, shall as a minimum, meet equivalent qualification criteria as specified for professional-technical personnel in Section 4.4 of ANSI N18.1-1971.

MEETING FREQUENCY

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

QUORUM

6.5.3.7 The minimum quorum of the PNSC necessary for the performance of the PNSC activities of the Technical Specifications shall consist of the PNSC Chairman or his designated alternate and three members including alternates.

ACTIVITIES

6.5.3.8 The PNSC activities shall include the following:

a. Review of all procedures required by Specification 6.8 and changes thereto and any other procedures and changes thereto, any of which constitute an unreviewed safety question or involve a change to the Technical Specifications.

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* Or designated alternate.

ACTIVITIES (Continued)

b. Review of all proposed tests or experiments that constitute an unreviewed safety question or involve a change to the Technical Specifications.

c. Review of all proposed modifications that constitute an unreviewed safety question or involve a change to the Technical Sepcifications.

d. Review of all proposed changes to the Technical Specifications and Operating License.

e. Review of reports on violations of Technical Specifications.

f. Performance of special reviews, investigations (or analyses), and reports thereon as requested by the Manager - Corporate Nuclear Safety Section.

g. Review of events requiring 24 hours written notification to the Commission.

h. Review of facility operations to detect potential nuclear safety hazards.

i. Annual review of the Security Plan.

j. Annual review of the Emergency Plan.

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section.

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

AUTHORITY

6.5.3.9 The PNSC shall provide written notification within 24 hours to the Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section of disagreement between recommendations of the PNSC and the actions contemplated by the Plant General Manager; however, the course determined by the Plant General Manager shall be followed.

RECORDS

6.5.3.10 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications. Copies shall be provided to the Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section.

6.5.4 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.4.1 The Corporate Nuclear Safety Section (CNSS) of the Corporate Nuclear Safety & Research Department shall function to provide independent review of significant plant changes, tests, and procedures; verify that REPORTABLE OCCURRENCES are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer.

ORGANIZATION

6.5.4.2 The individuals assigned responsibility for independent reviews shall be specified in technical disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. administrative controls

i. seismic and environmental

j. quality assurance practices

6.5.4.3 The Manager - Corporate Nuclear Safety Section shall have an academic degree in an engineering or related field and, in addition, shall have a minimum of ten years related experience, of which a minimum of five years shall be in the operation and/or design of nuclear power plants.

ORGANIZATION (Continued)

6.5.4.4 The independent safety review program reviewers shall have an academic degree in an engineering or related field or equivalent and, in addition, shall have a minimum of five years related experience.

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

6.5.4.6 At least three individuals, qualified as discussed in 6.5.4.4 above shall review each item submitted under the requirements of Section 6.5.4.9.

6.5.4.7 Independent safety reviews shall be performed by individuals not directly involved with the activity under review or responsible for the activity under review.

6.5.4.8 The Corporate Nuclear Safety Section independent safety review program shall be conducted in accordance with written, approved procedures.

REVIEW

6.5.4.9 The Corporate Nuclear Safety Section shall perform reviews of the following:

a. All procedures required by Specification 6.8 and changes thereto that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications.

b. All proposed tests or experiments that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications.

c. All proposed modifications that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications.

d. All procedures required by Specification 6.8 and changes thereto and all proposed tests or experiments that constitute a change to the safety analysis report. This review may be performed after appropriate management approval; implementation may proceed prior to completion of the review.

e. All proposed modifications that constitute a change to the facility as described in the safety analysis report. This review may be performed after appropriate management approval; implementation may proceed prior to completion of the review.

ORGANIZATION (Continued)

f. All proposed changes to the Technical Specifications and Operating License.

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

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

i. Events requiring 24 hour written notification to the Commission.

j. Reports and minutes of the PNSC.

k. Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager - Corporate Nuclear Safety Section by the on-site operating organization or other functional organizational units within Carolina Power & Light Company.

6.5.4.10 Review of items considered under 6.5.4.9(g) through (i) above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.4.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNS files.

b. Recommendations and concerns shall be submitted to the Plant General Manager and Vice President - Brunswick Nuclear Projects, within 14 days of completion of the review.

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

6.5.5 CORPORATE QUALITY ASSURANCE AUDIT PROGRAM

FUNCTION

6.5.5.1 The Performance Evaluation Unit (PEU) of the Corporate Quality Assurance Department shall function to perform audits of facility activities specified in Specification 6.5.5.2.

AUDITS

6.5.5.2 Audits of facility activities shall be performed by the PEU. These audits shall encompass:

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

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

c. The results of actions taken to correct deficienies occurring in facility equipment, structures, systems, or methods 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 12 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. The radiological environmental monitoring program and the results thereof at least once per 12 months.

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

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

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

AUDITS (Continued)

l. Any other area of facility operation considered appropriate by the Corporate Quality Assurance Performance Evaluation Unit.

6.5.5.3 Personnel performing the quality assurance audits shall have access to the plant operating records.

RECORDS

6.5.5.4 Records of audits shall be prepared and retained.

6.5.5.5 Audit reports encompassed by 6.5.5.2 above shall be prepared, approved by the Principal QA Specialist - Performance Evaluation Unit, and forwarded to the Executive Vice President - Power Supply and Engineering and Construction; Vice President - Brunswick Nuclear Projects; Vice President - Corporate Nuclear Safety and Research; Plant General Manager; and others, as appropriate, within 30 days after completion of the audit.

AUTHORITY

6.5.5.6 The Principal QA Specialist - Performance Evaluation Unit under the Manager - Corporate Quality Assurance shall be responsible for the following:

a. The administering of the Corporate Quality Assurance Audit Program.

b. The approval of the individual(s) selected to conduct quality assurance audits.

PERSONNEL

6.5.5.7 Audit personnel shall be independent of the area audited.

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

6.5.5.9 Qualified outside consultants or other individuals independent from those personnel directly involved in plant operation shall be used to augment the audit teams when necessary.

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

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

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM (Continued)

6.5.6.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 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 Plant General Manager and submitted to the Manager - Corporate Nuclear Safety and the Vice President - Brunswick Nuclear Projects.

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 NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant General Manager. 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 Vice President - Brunswick Nuclear Projects, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

SAFETY LIMIT VIOLATION (Continued)

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. OFFSITE DOSE CALCULATION MANUAL implementation.

h. PROCESS CONTROL PROGRAM implementation.

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

6.9 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 Regional Administrator, United States Nuclear Regulatory Commission, Region II, unless otherwise noted.

STARTUP REPORTS

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

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

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all

STARTUP REPORTS (Continued)

three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS 1/

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

PERSONNEL EXPOSURE AND MONITORING REPORT 2/

6.9.1.5 Reports required on an annual basis shall include a tabulation of the number of station, utility, and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated, based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources shall be assigned to specific major work functions.

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

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT [3/]

6.9.1.6 Routine radiological environmental operating reports covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.1.7 The Annual Radiological Environmental Operating Report shall include the following:

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

b. Results of land uses censuses required by Specification 3.12.2.

c. Results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of Table 3 in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data should be submitted as soon as possible in a supplementary report.

d. A summary description of the radiological environmental monitoring program.

e. At least two legible maps [4/] covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor.

f. The results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

g. Discussion of all deviations from the sampling schedule of Table 3.12.1-1.

h. Discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

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

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT [5/]

6.9.1.8 Routine radioactive effluent release reports covering the operation of the facility during the previous 6 months of operation shall be submitted within the time periods specified in Specifications 6.9.1.9 and 6.9.1.10 below.

6.9.1.9 The portion of the Semiannual Radioactive Effluent Release Reports to be submitted within 60 days after January 1 and July 1 of each year shall include the following:

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

b. The information specified below for each class of solid waste (as defined by 10 CFR Part 61, when implemented) shipped offsite during the report period:

   1. Container volume,

   2. Total curie quantity (specifiy whether determined by measurement or estimate),

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

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

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

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

c. 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.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

d. Any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) or 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 3.12.2.

6.9.1.10 The portion of the Semiannual Radioactive Effluent Release Report to be submitted within 90 days after January 1 of each year shall include the following:

a. An annual summary of hourly metorological data collected over the previous calendar year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed wind direction, and atmospheric stability. 6/

b. As assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the station during the previous calendar year.

MONTHLY OPERATING REPORTS

6.9.1.11 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the U. S. Nuclear Regulatory Commission, Region II, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.12 The REPORTABLE OCCURRENCES of Specifications 6.9.1.13 and 6.9.1.14 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

6.9.1.13 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region II,

6/ In lieu of submission with the Semiannual 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.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of the 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.

Note: Instrument drift discovered as a result of testing need not be reported under this item (but see 6.9.1.13.e, 6.9.1.13.f, and 6.9.1.14.a below).

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.

Note: If specified action is taken when a system is found to be operating between the most conservative and least conservative aspects of a limiting condition for operation listed in the Technical Specifications, the limiting condition for operation is not considered to have been violated and no report need be submitted under this section (but see 6.9.1.14.b below).

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

Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this section.

d. Reactivity anomalies involving disagreement with 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.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

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

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

Note: For 6.9.1.13.e and 6.9.1.13.f, reduced redundancy that does not result in loss of system function need not be reported under this section (but see 6.9.1.14.b and 6.9.1.14.c below).

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.

Note: This item is intended to provide for reporting of potentially generic problems.

THIRTY DAY WRITTEN REPORTS

6.9.1.14 The types of events listed below shall be the subject of written reports to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region II within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of the 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.

THIRTY DAY WRITTEN REPORTS (Continued)

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 (but see 6.9.1.13.a and 6.9.1.13.b above).

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 (but see 6.9.1.13.b above).

Note: Routine surveillance testing, instrument calibration, or preventive maintenance that require configurations described in 6.9.1.14.a and 6.9.1.14.b above need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems (but see 6.9.1.13.f above).

d. Abnormal degradation of systems other than those specified in 6.9.1.13.c above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region II within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Reactor coolant specific activity analysis, Specification 3.4.5.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

SPECIAL REPORTS (Continued)

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire barrier penetation, Specification 3.7.8.

h. Liquid Effluents Dose, Specification 3.11.1.2.

i. Liquid Radwaste Treatment, Specification 3.11.1.3.

j. Dose - Noble Gases, Specification 3.11.2.2.

k. Dose - Iodine-131, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

l. Gaseous Radwaste Treatment, Specification 3.11.2.4.

m. Ventilation Exhaust Treatment, Specification 3.11.2.5.

n. Total Dose, Specification 3.11.4.

o. Monitoring Program, Specification 3.12.1.b.

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 OCCURRENCES 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 detector 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 inservice 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 (1) meetings of the PNSC, (2) meetings of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC), and (3) the independent reviews performed by the Corporate Nuclear Safety Section.

m. Records for Environmental Qualification which are covered under the provisions of paragraph 6.16.

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

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "Control Device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 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 (RWP)*. 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 Radiation 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 Operations Shift Foreman on duty and/or the qualified plant health physicist.

* Health Physics personnel or personnel escorted by 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 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the ODCM:

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

   1. Sufficiently detailed information to totally support 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);

   2. A determinatin that the changes will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and,

   3. Documentatin of the fact that the change has been reviewed and found acceptable by the PNSC.

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

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

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

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

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

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

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

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEM[7/]

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

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

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

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

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

4. An evaluation of the change that shows the predicted release of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change 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 amendments thereto;

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

7. An estimte of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable to the PNSC.

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

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

6.16 ENVIRONMENTAL QUALIFICATION

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

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2
RADIOLOGICAL EFFLUENT TECHNCIAL SPECIFICATIONS
(RETS)

DEFINITIONS

SECTION

1.0 DEFINITIONS

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SECTION

1.0 DEFINITIONS (Continued)

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ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

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.

ACTION

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

AVERAGE PLANAR EXPOSURE

The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all of the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

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 indication 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 simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CHANNEL FUNCTIONAL TEST (Continued)

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

CORE ALTERATION

CORE ALTERATION shall be the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls in the reactor core with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe, conservative location.

CRITICAL POWER RATIO

The CRITICAL POWER RATIO (CPR) shall be ratio of that power in the assembly which is calculated, by application of the GEXL correlation, to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be concentration of I-131, μCi/gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The following is defined equivalent to 1 μCi of I-131 as determined from Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites": I-132, 28 μCi; I-133, 3.7 μCi; I-134, 59 μCi; I-135, 12 μCi.

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes with half lives greater than 15 minutes making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to recirculation pump breaker trip from initial movement of the associated:

a. Turbine stop valves, and

b. Turbine control valves.

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 off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting and sequence loading delays where applicable.

LIMITING CONTROL ROD PATTERN

A LIMITING CONTROL ROD PATTERN shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE

LINEAR HEAT GENERATION RATE (LHGR) shall be the power generation in an arbitrary length of fuel rod, usually one foot. It is the integral of the heat flux over the heat transfer area associated with the unit length, usually measured in kW/ft.

LOGIC SYSTEM FUNCTIONAL TEST

A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all relays and contacts of a logic circuit, from sensor output to activated device, to ensure that components are OPERABLE.

MAXIMUM FRACTION OF LIMITING POWER DENSITY

MAXIMUM FRACTION OF LIMITING POWER DENSITY shall be the highest value of LINEAR HEAT GENERATION RATE (LHGR) divided by the corresponding LHGR limit occurring in the reactor core.

MAXIMUM TOTAL PEAKING FACTOR

The MAXIMUM TOTAL PEAKING FACTOR (MTPF) shall be the largest TPF which exists in the core for a given class of fuel for a given operating condition.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does not include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

ODYN OPTION A

ODYN OPTION A shall be analyses which refer to minimum critical power ratio limits which are determined using a transient analysis plus an analysis uncertainty penalty.

ODYN OPTION B

ODYN OPTION B shall be analyses which refer to minimum critical power ratio limits determined using a transient analysis which includes a requirement for 20% scram insertion times to reduce the analysis uncertainty penalty.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATIONAL 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.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and are 1) described in Section 13 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.1, or

b. All equipment hatches are closed and sealed.

c. Each containment air lock is OPERABLE pursuant to Specification 3.6.1.3.

d. The containment leakage rates are within the limits of Specification 3.6.1.2.

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

PROCESS CONTROL PROGRAM (PCP)

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.

REACTOR PROTECTION SYSTEM RESPONSE TIME

REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids.

REFERENCE LEVEL ZERO

The REFERENCE LEVEL ZERO point is arbitrarily set at 367 inches above the vessel zero point. This REFERENCE LEVEL ZERO is approximately mid-point on the top fuel guide and is the single reference for all specifications of vessel water level.

REPORTABLE OCCURRENCE

A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.13 and 6.9.1.14.

ROD DENSITY

ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of notches. All rods fully inserted are equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGERITY shall exist when:

a. All automatic Reactor Building ventilation system isolation valves or dampers are OPERABLE or secured in the isolated position.

b. The standby gas treatment system is OPERABLE pursuant to Specification 3.6.6.1.

c. At least one door in each access to the Reactor Building is closed.

d. The sealing mechanism associated with each penetration (e.g., welds bellows, or O-rings) is OPERABLE.

SHUTDOWN MARGIN

SHUTDOWN MARGIN shall be the amount of reactivity by which the reactor would be subcritical assuing that all control rods capable of insertion are fully inserted except for the analytically determined highest worth rod which is assumed to be fully withdrawn, and the reactor is in the shutdown condition, cold, 68°F, and Xenon-free.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor 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.

SPIRAL RELOAD

A SPIRAL RELOAD is the reverse of a SPIRAL UNLOAD. Except for two diagonal fuel bundles around each of the four SRMs, the fuel in the interior of the core, symmetric to the SRMs, is loaded first.

SPIRAL UNLOAD

A SPIRAL UNLOAD is a core unload performed by first removing the fuel from the outermost control cells (four bundles surrounding a control blade). Unloading continues in a spiral fashion by removing fuel from the outermost periphery to the interior of the core, symmetric about the SRMs, except for two diagonal fuel bundles around each of the four SRMs.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

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

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

THEMAL POWER

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

TOTAL PEAKING FACTOR

The TOTAL PEAKING FACTOR (TPF) shall be the ratio of local LHGR for any specific location on a fuel rod divided by the average LHGR associated with the fuel bundles of the same type operating at the core average bundle power.

UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE.

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

VENTILATION EXHAUST TREATMENT SYSTEM (Continued)

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. 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.
P	Prior to each release.
NA	Not applicable.

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	$> 212^0F$
4. COLD SHUTDOWN	Shutdown	$\leq 212^0F$
5. REFUELING*	Refuel**	$\leq 212^0F$

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

**See Special Test Exception 3.10.3.

***See Special Test Exception 3.10.1.

TABLE 3.3.5.7-1 (Continued)

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE FLAME	HEAT	SMOKE
3. Diesel Generator Building (Cont'd)			
Zone 7 23'	0	0	5
Zone 8 23'	0	0	5
Zone 9 23'	0	0	8
Zone 10 50'	0	0	9
4. Service Water Buliding			
Zone 1 4'	0	0	7
Zone 2 20'	0	0	6
5. AOG Building			
Zone 1 20'	0	0	2
Zone 2 20'	0	0	2
Zone 3 20'	1	5	1
Zone 4 37' - 49'	1	6	6

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3.5.8-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.8-1.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint 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 inoperable, 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 3.3.5.8-1. 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 6.9.1.14.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.8 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3.5.8-1.

NOTE: See Bases 3/4.3.5.8.

TABLE 3.3.5.8-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT[a]	APPLICABILITY	ACTION
1. Liquid Radwaste Radioactivity Effluent Monitor (Providing alarm and automatic termination of release)	*	110
2. Liquid Radwaste Effluent Flow Measurement Device	*	111
3. Main Service Water Effluent Radioactivity Monitor	*	112
4. Stabilization Pond Effluent Composite Sampler	**	113
5. Stabilization Pond Effluent Flow Measurement Device	**	114
6. Condensate Storage Tank Level Indicating Device	*	115
7. Service Water Effluent from Augmented Off-Gas Precooler Radioactivity Monitor	***	112
8. Reactor Building Component Cooling Water (Service Water) Radioactivity Monitors		
a. Effluent from Residual Heat Removal Heat Exchanger A	****	112
b. Effluent from Residual Heat Removal Heat Exchanger B	****	112
c. Effluent from Reactor Building Closed Cooling Water Heat Exchangers	****	112
d. Effluent from Division I Residual Heat Removal Pump Seal Coolers	****	112
e. Effluent from Division II Residual Heat Removal Pump Seal Coolers	****	112

TABLE 3.3.5.8-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

ACTIONS

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 4.11.1.1.2, 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 4.11.1-1. Otherwise, suspend releases via this pathway.

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 3.3.5.8-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

NOTES

* At all times

** At all times other than when the line is valved out and locked. [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].

**** At all times once these monitors are installed and become fully operational. [NOTE: These monitors are to be installed pending completion of future plant modifications.]

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

TABLE 4.3.5.8-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT[a]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1.	Liquid Radwaste Radioactivity Effluent Monitor (Providing alarm and automatic termination of release)	D	M	R[b]	Q[c]
2.	Liquid Radwaste Effluent Flow Measurement Device	D[e]	NA	R	Q
3.	Main Service Water Effluent Radioactivity Monitor	D	M	R[b]	Q[d]
4.	Stabilization Pond Effluent Composite Sampler	D	NA	R	Q
5.	Stabilization Pond Effluent Flow Measurement Device	D	NA	R	Q
6.	Condensate Storage Tank Level Indicating Device	D[f]	NA	R	Q
7.	Service Water Effluent from Augmented Off-Gas Precooler Radioactivity Monitor	D	M	R[b]	Q
8.	Reactor Building Component Cooling Water (Service Water) Radioactivity Monitors				
	a. Effluent from Residual Heat Removal Heat Exchanger A	D	M	R[b]	Q[d]
	b. Effluent from Residual Heat Removal Heat Exchanger B	D	M	R[b]	Q[d]
	c. Effluent from Reactor Building Closed Cooling Water Heat Exchangers	D	M	R[b]	Q[d]
	d. Effluent from Division I Residual Heat Removal Pump Seal Coolers	D	M	R[b]	Q[d]
	e. Effluent from Division II Residual Heat Removal Pump Seal Coolers	D	M	R[b]	Q[d]

TABLE 4.3.5.8-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

(b) 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.

(c) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument controls not set in "operate" mode.

(d) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument controls not set in "operate" mode.

(e) 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.

(f) During liquid additions to the tank.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5.9 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3.5.9-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The setpoints shall be determined in accordance with the methodology as described in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: As shown in Table 3.3.5.9-1.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, 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 monitoring instrumentation channel OPERABLE, take the ACTION shown in Table 3.3.5.9-1. 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 6.9.1.14.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.5.9 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3.5.9-1.

NOTE: See Bases 3/4.3.5.9.

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT[(a)]	APPLICABILITY	ACTION
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM (Continued)		
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 and automatic isolation)	**	121
5. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM MONITOR (Downstream of AOG Treatment System)		
a. Noble Gas Activity Monitor (providing alarm)	***	123
6. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM		
a. Recombiner Train A		
1. 1st Hydrogen Monitor	****	125
2. 2nd Hydrogen Monitor	****	125
b. Recombiner Train A		
1. 1st Hydrogen Monitor	****	125
2. 2nd Hydrogen Monitor	****	125
7. HOT SHOP VENTILATION MONITORING SYSTEM		
a. Iodine Sampler Cartridge	*	127
b. Particulate Sampler Filter	*	127

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

ACTIONS

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

a. The GASEOUS RADWASTE TREATMENT SYSTEM is not bypassed [Prior to the Augmented Off-Gas Treatment System becoming operational, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute offgas holdup line including stack filtration], 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 12 hours and these samples are analyzed for gross noble gas activity within 24 hours.

ACTION 125 - With less than two channels OPERABLE in the operating recombiner train, operation of the train may continue provided proper function of the recombiner is assured by monitoring recombiner temperature in accordance with approved procedures.

With less than one channel OPERABLE in the operating recombiner train, operation of the train may continue provided grab samples from the 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.

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 4.11.2-1.

TABLE 3.3.5.9-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT[a]	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
3.	TURBINE BUILDING VENTILATION MONITORING SYSTEM		
	a. Noble Gas Activity Monitor	*	123
	b. Iodine Sampler Cartridge	*	127
	c. Particulate Sampler Filter	*	127

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

NOTES

* At all times.

** During operation of the main condenser air ejector.

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

**** At all times during recombiner train operation.

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

TABLE 4.3.5.9-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT[a]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. MAIN STACK MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R[b]	Q[d]	*
b. Iodine Sampler Cartridge	W	NA	NA	NA	*
c. Particulate Sampler Filter	W	NA	NA	NA	*
d. System Effluent Flow Rate Measurement Device	D	NA	R	Q	*
e. Sampler Flow Rate Measurement Device	D	NA	R	Q	*
2. REACTOR BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R[b]	Q[d]	*
b. Iodine Sampler Cartridge	W	NA	NA	NA	*
c. Particulate Sampler Filter	W	NA	NA	NA	*
d. System Effluent Flow Rate Measurement Device	D	NA	R	Q	*
e. Sampler Flow Rate Measurement Device	D	NA	R	Q	*

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT[(a)]	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. TURBINE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor	D	M	R[(b)]	Q[(d)]	*
b. Iodine Sampler Cartridge	W	NA	NA	NA	*
c. Particulate Sampler Filter	W	NA	NA	NA	*
d. System Effluent Flow Rate Measurement Device	D	NA	R	Q	*
e. Sampler Flow Rate Measurement Device	D	NA	R	Q	*
4. MAIN CONDENSER AIR EJECTOR RADIOACTIVITY MONITOR (Prior to input to treatment system)					
a. Noble Gas Activity Monitor (Providing alarm and automatic isolation)	D	M	R[(b)]	Q[(c)]	**
5. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM MONITOR (Downstream of AOG Treatment System)[(f)]					
a. Noble Gas Activity Monitor (Providing alarm)	D	P	R[(b)]	Q	***

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT(a)	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. MAIN CONDENSER OFF-GAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM(f)					
a. Recombiner Train A					
1. 1st Hydrogen Monitor	D	NA	Q(e)	M	****
2. 2nd Hydrogen Monitor	D	NA	Q(e)	M	****
b. Recombiner Train B					
1. 1st Hydrogen Monitor	D	NA	Q(e)	M	****
2. 2nd Hydrogen Monitor	D	NA	Q(e)	M	****
7. HOT SHOP VENTILATION MONITORING SYSTEM					
a. Iodine Sampler Cartridge	W	NA	NA	NA	*
b. Particulate Sampler Filter	W	NA	NA	NA	*

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

(b) 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.

(c) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument not set in "operate" mode.

(d) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure (High-voltage low).
3. Instrument indicates a downscale failure.
4. Instrument not set in "operate" mode.

(e) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

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

TABLE 4.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES

* At all times other than when the line is valved out and locked.

** During operation of the main condenser air ejector.

*** At all times other than when the line is valved out and locked (once the Augmented Off-Gas Treatment System becomes fully operational).

**** During recombiner train operation.

3/4.3.6 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUEMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation trip systems shown in Table 3.3.6.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.1-2.

APPLICABILITY: CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6.1-2, declare the trip system inoperable until the trip system is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the requirements for the Minimum Number of OPERABLE Trip Systems per Operating Pump not satisfied for one Trip Function, restore the inoperable trip system to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.3.6.1.1 Each ATWS recirculation pump trip system instrumentation trip system shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.1.1-1.

4.3.6..2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

TABLE 3.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION AND INSTRUMENT NUMBER	MINIMUM NUMBER OPERABLE TRIP SYSTEMS PER OPERATING PUMP
1. Reactor Vessel Water Level - Low, Level 2 (B21-LT-N024A-2,B-2 and B21-LT-N025A-2,B-2) (B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	1
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	1

TABLE 3.3.6.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION AND INSTRUMENT NUMBER	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Level 2 (B21-LTM-N024A-2,B-2; B21-LTM-N025A-2,B-2)	$\geq$ + 112 inches*	$\geq$ + 112 inches*
2. Reactor Vessel Pressure - High (B21-PS-N045A,B,C,D)	$\leq$ 1120 psig	$\leq$ 1120 psig

*Vessel water levels refer to REFERENCE LEVEL ZERO.

TABLE 4.3.6.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION AND INSTRUMENT NUMBER	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low, Level 2			
(B21-LT-N024A-2,B-2 and B21-LT-N025A-2,B-2)	NA[(a)]	NA	R[(b)]
(B21-LTM-N024A-2,B-2 and B21-LTM-N025A-2,B-2)	D	M	M
2. Reactor Vessel Pressure - High (B21-PS-N045A, B, C, D)	NA	M	R

(a) The transmitter channel check is satisfied by the trip unit channel check. A separate transmitter check is not required.

(b) Transmitters are exempted from the monthly channel calibration.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.6.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.6.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values Column of Table 3.3.6.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within one hour.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within one hour.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

SURVEILLANCE REQUIREMENTS

4.3.6.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.6.2.1-1.

4.3.6.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months and shall include calibration of time delay relays and timers necessary for proper functioning of the trip system.

4.3.6.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of both trip systems shown in Table 3.3.6.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure, or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months.

TABLE 3.3.6.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Stop Valve - Closure (EHC-SVOS-1X, 2X, 3X, 4X)	2[b]
2. Turbine Control Valve - Fast Closure (EHC-PSL-1756, 1757, 1758, 1759)	2[b]

[a] A trip system may be placed in an inoperable status for up to 2 hours for required surveillance, provided that the other trip system in OPERABLE.

[b] These functions are bypassed when turbine first stage pressure is equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

TABLE 3.3.6.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Stop Valve-Closure (EHC-SVOS-1X, 2X, 3X, 4X)	$\leq$ 10% closed	$\leq$ 10% closed
2. Turbine Control Valve-Fast Closure (EHC-PSL-1756,1757,1758,1759)	$\geq$ 500 psig	$\geq$ 500 psig

TABLE 3.3.6.2-3

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)
1. Turbine Stop Valve-Closure (EHC-SVOS-1X, 2X, 3X, 4X)	≤ 0.175
2. Turbine Control Valve-Fast Closure (EHC-PSL-1756, 1757, 1758, 1759)	≤ 0.175

TABLE 4.3.6.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure (EHC-SVOS-1X, 2X, 3X, 4X)	M*	R
2. Turbine Control Valve-Fast Closure (EHC-PSL-1756, 1757, 1758, 1759)	M*	R

*Including trip system logic testing.

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4. The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In CONDITIONS 1, 2, and 3:

1. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1, but less than 10 µmho/cm at 25°C and less than 0.5 ppm, respectively, operation may continue for up to 24 hours and this condition need not be reported to the Commission per Specification 6.9.1.12, provided that operation under these conditions shall not exceed 336 hours per year. The provisions of Specification 3.0.4 are not applicable.

2. With the conductivity or chloride concentration exceeding the limits specified in Table 3.4.4-1 for more than 24 hours during one continuous time interval or with the conductivity exceeding 10 µmho/cm at 25°C or chloride exceeding 0.5 ppm, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. At all other times with the conductivity and/or chloride concentration of the reactor coolant in excess of the limit specified in Table 3.4.4-1, restore the conductivity and/or chloride concentration to within the limit within 48 hours.

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the reactor coolant shall be limited to:

a. ≤ 0.2 μCi/gram DOSE EQUIVALENT I-131, and

b. ≤ $100/\bar{E}$ Ci/gram.

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

ACTION:

a. In CONDITIONS 1, 2, and 3, with the specific activity of the reactor coolant:

1. > 0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤ 4.0 μCi/gram, operation may continue for up to 48 hours provided that operation under these conditions shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

2. > 0.2 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or > 4.0 μCi/gram, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

3. > $100/\bar{E}$ μ Ci/gram, be in at le[illegible] SHUTDOWN with the main steam line isolation valves clo[illegible] hours and in COLD SHUTDOWN within the next 24 hours.

b. In CONDITIONS 1, 2, 3, or 4,

1. With the specific activity of the primary coolant > 0.2 μCi/gram DOSE EQUIVALENT I-131 or > $100/\bar{E}$ μCi/gram, perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 at least once per 4 hours until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE report shall be prepared and submitted to the Commission pursuant to Specification 6.9.1.12. This report shall contain the results of the specific activity analyses and the time duration when the specific activity coolant exceeded 0.2 μCi/gram DOSE EQUIVALENT I-131 together with the below additional information.

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.4 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.4.

APPLICABILITY: CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 212°F.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The structural integrity of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of the accessible interior and exterior surfaces of the containment and verifying no apparent changes in appearance of the surfaces or other abnormal degradation. Any abnormal degradation of the primary containment detected during the required inspections shall be reported to the Commission pursuant to Specification 6.9.1.12.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (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

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1-1. If the stabilization pond or service water samples analyzed according to Table 4.11.1-1 indicate concentrations of any gamma-emitting radionuclides greater than 5×10^{-6} μCi/ml (trigger level), then the liquid wastes exceeding the trigger level shall be sampled and analyzed according to the sampling and analysis program of Table 4.11.1-2 until such time as the sample concentration of each gamma-emitting nuclide is less than 5×10^{-6} μCi/ml.

4.11.1.1.2 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 3.11.1.1.

NOTE: See Bases 3/4.11.1.1

TABLE 4.11.1-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml) (a)(e)
A.1. 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)
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (gamma emitters)	1×10^{-5}
2. Circulating Water Pit	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(d)	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(d) (Potential Continuous Release)	W During System Operation	W During System Operation	Principal Gamma Emitters(g)	5×10^{-7} (b)

TABLE 4.11.1-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

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 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) The stabilization pond and service water liquid release types represent potential release pathways and not actual release pathways. Surveillance of these pathways is intended to alert the plant to a potential problem; analysis for principal gamma emitters should be sufficient to meet this intent. If analysis for principal gamma emitters indicates a problem (i.e., exceeds the trigger level of 5×10^{-6} µCi/ml), then complete sampling and analyses shall be performed as per Table 4.11.1-2.

(e) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\ \sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

where:

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

$\sigma_b = (N/t_b)^{1/2}$

= standard deviation of background (cpm)

TABLE 4.11.1-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

N = background count rate (cpm)

t_b = time background counted for (min)

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

(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 composite sampling instrumentation becomes available and is OPERABLE, daily grab sampling of the stabilization pond effluent will not be required during release and the composite sample will be analyzed on a weekly basis.

(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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

(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 4.11.1-1.

TABLE 4.11.1-2

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
FOR POTENTIAL RELEASE PATHWAYS WHICH HAVE EXCEEDED TRIGGER LEVELS

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

TABLE 4.11.1-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
FOR POTENTIAL RELEASE PATHWAYS WHICH HAVE EXCEEDED TRIGGER LEVELS

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 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) Until such time as continuous proportional composite samplers are installed on the service water discharge line, daily grab sampling of the service water effluent will be required for use in making up the composite.

(e) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

Where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

N = background count rate (cpm)

t_b = time background counted for (min)

TABLE 4.11.1-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
FOR POTENTIAL RELEASE PATHWAYS WHICH HAVE EXCEEDED TRIGGER LEVELS

E	=	counting efficiency, as counts per disintegration
V	=	volume or mass of sample
2.22	=	conversion factor (dpm/pCi)
Y	=	fractional radiochemical yield
λ_i	=	radioactive decay constant of ith nuclide (sec^{-1})
t_e	=	elapsed time between sample collection and counting (sec)

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

(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 composite sampling instrumentation becomes available and is OPERABLE, daily grab sampling of the stabilization pond effluent will not be required during release and the composite sample will be analyzed on a weekly basis.

(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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

(h) A continuous release is the discharge of liquid waste of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

RADIOACTIVE EFFLUENTS

DOSE - LIQUID EFFLUENTS

LIMITING CONDITION FOR OPERATION

3.1 .1.2 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 less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ, and

b. During any calendar year to less than or equal to 6 mrem to the total body and to less than or equal to 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 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(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

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

NOTE: See Bases 3/4.11.1.2

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 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 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 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 Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3 Doses due to liquid releases from the site to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the ODCM.

NOTE: See Bases 3/4.11.1.3

RADIOACTIVE EFFLUENTS

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

3.11.1.4 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	10 Ci
b. 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 above limit, without delay suspend all addition of radioactive material to the tank, within 48 hours reduce the tank's contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the tanks listed shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

NOTE: See Bases 3.4.11.1.4

TABLE 3.3.5.9-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

NOTES

* At all times.

** During operation of the main condenser air ejector.

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

**** At all times during recombiner train operation.

(a) Refer to Appendix E of the OFFSITE DOSE CALCULATION MANUAL for specific instrumentation identification numbers.

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site 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, tritium, and 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).

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 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 4.11.2-1.

NOTE: See Bases 3/4.11.2.1

TABLE 4.11.2-1

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 Emmitters (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(h)	M(c)(d) Grab Sample	M(c)	Principal Gamma Emmitters (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 Emmitters (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 4.11.2-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

TABLE NOTATION

(a) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\sigma_b}{E\ V\ 2.22\ Y\ \exp(-\lambda_i t_e)}$$

Where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

N = background count rate (cpm)

t_b = time background counted for (min)

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement.

(b) The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for

TABLE 4.11.2-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.8.

(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 spent 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. Spent 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 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(f) Sample 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.

(h) Monthly grab samples to be analyzed for principal gamma emitters and tritium are not applicable for the Hot Shop environmental release point. In addition, the Hot Shop release point does not have a continuous noble gas monitor and, therefore, the noble gas activity analysis requirements of Table 4.11.2-1 are not applicable.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 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: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation;

b. During any calendar year: Less than or equal to 20 mrad for gamma radiation and less than or equal to 40 mrad for beta radiation.

APPLICABILITY: At all times.

ACTIONS:

a. With the calculated air dose from radioactive noble gases 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(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

4.11.2.2 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.

NOTE: See Bases 3/4.11.2.2

DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 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

4.11.2.3 Dose Calculations - Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-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.

NOTE: See Bases 3/4.11.2.3

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM shall be in operation.

APPLICABILITY: Whenever the main condenser air ejector (evacuation) system is in operation.

ACTION:

a. With gaseous radwaste from the main condenser air ejector system being discharged without treatment for more than 7 days, 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. Identification of the inoperable equipment or subsystems and the reason for 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

4.11.2.4 The readings of the relevant instruments shall be checked at least once per 12 hours when the main condenser air ejector is in use to ensure that the GASEOUS RADWASTE TREATMENT SYSTEM is functioning.

NOTE: See Bases 3/4.11.2.4

RADIOACTIVE EFFLUENTS

VENTILATION EXHAUST TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.5 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 other than when the VENTILATION EXHAUST TREATMENT SYSTEM is undergoing routine maintenance.

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. 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

4.11.2.5 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM, when the VENTILATION EXHAUST TREATMENT SYSTEM is not in use.

NOTE: See Bases 3/4.11.2.5

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: Whenever the main condenser air ejector system is in operation.*

ACTION:

a. With the concentration of hydrogen in the main condenser offgas 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

4.11.2.6 The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within the above limit by continuously monitoring the waste gases in the main condenser offgas treatment system with the hydrogen monitors required OPERABLE by Table 3.3.5.9-1 of Specification 3.3.5.9.

NOTE: See Bases 3/4.11.2.6

*This specification shall become applicable when the offgas recombiners become operational.

MAIN CONDENSER AIR EJECTOR RADIOACTIVITY RELEASE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.7 The release rate of the sum of the activities from the noble gases measured at the main condenser air ejector shall be limited to less than equal to 243,600 microcuries/second (the Kr-85m, 87, 88 and Xe-133, 135, 138 contribution after 30 minutes decay).

APPLICABILITY: During operation of the main condenser air ejector.

ACTION:

With the release rate of the sum of the activities from the 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

4.11.2.7.1 The radioactivity rate of noble gases at (near) the outlet of the main condenser air ejector shall be continuously monitored in accordance with Specification 3.11.2.1.

4.11.2.7.2 The release rate of the sum of the activities from the 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, or

b. Within 31 days following each refueling/maintenance outage, and

c. Within 4 hours following an increase, as indicated by the Condenser Air Ejector Noble Gas Activity Monitor, of 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.

NOTE: See Bases 3/4.11.2.7

RADIOACTIVE EFFLUENTS

DRYWELL VENTING OR PURGING

LIMITING CONDITION FOR OPERATION

3.11.2.8 The drywell shall be purged to the environment at a rate in conformance with Specification 3.11.2.1.

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

4.11.2.8 A sample analysis, as defined in Table 4.11.2-1, shall be performed prior to each drywell PURGE.

NOTE: See Bases 3/4.11.2.8

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 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 6.9.1.14.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 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. I. 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 6.14, to assure SOLIDIFICATION of subsequent batches of waste.

NOTE: See Bases 3/4.11.3

3/4.11.4 TOTAL DOSE (40 CFR PART 190)

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC, due to releases of radioactivity and 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 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made which, in addition to doses due to effluents, include direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, 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.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE (40 CFR PART 190)

SURVEILLANCE REQUREMENTS

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

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

NOTES: See Bases 3/4.11.4

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.6, 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 sampling medium at a specific location exceeding the reporting levels of Table 3.12.1-2 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 3.12.1-2 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 3.11.1.2, 3.11.2.2, and 3.11.2.3. The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report. 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 Operating Report.

c. With milk or fresh leafy vegetables unavailable from one or more of the sample locations required by Table 3.12.1-1, identify locations for obtaining replacement samples and add them to the radiological

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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 6.9.1.8, 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

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12.1-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 3.12.1-1 and the detection capabilities required by Table 4.12.1-1.

NOTE: See Bases 3/4.12.1

TABLE 3.12.1-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency Of Analysis
1. DIRECT RADIATION[b]	40 Locations. At each location with 2 or more dosimeters or one or more instruments for continuously measuring and recording dose rate, placed as follows: An inner ring of stations, with at least one in each meteorological sector in the general area of the site boundary as is reasonably accessible and practical; An outer ring of stations, with at least one in each meteorological sector at distances of 8 km or greater from the site as is reasonably accessible and practical; and The balance of stations to be placed in special interest areas such as population centers, nearby residences, schools, and in at least one or two areas to serve as control stations.	Q	Gamma Dose - Q

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[(a)]	Sampling and Collection Frequency	Type and Frequency Of Analysis
2. AIRBORNE - Radioiodine and Particulate	5 Locations, as follows: 3 samples from different sectors as close to the SITE BOUNDARY as is reasonably accessible, one of which being at the highest calculated annual average ground level D/Q; 1 sample from the vicinity of a nearby community; and 1 sample from a control location, as for example greater than 15 km distant and in a less prevalent wind direction[(c)]	Continuous sampler operation with sample collection weekly or as required by dust loading, whichever is more frequent.	Radioiodine Cannister: I-131 analysis - W Particulate sampler: Gross beta radioactivity analysis following filter change;[(d)] Gamma isotopic analysis[(e)] of composite (by location) - Q
3. WATERBORNE a. Surface[(f)]	2 locations, as follows: 1 sample upstream 1 sample downstream	Composite[(g)] sample collection - M	Gamma Isotopic Analysis[(e)] - M Tritium Analysis - Q

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE (Continued)			
b. Sediment from shoreline	1 location with a sample taken from a downstream area with existing or potential recreational value	SA	Gamma Isotopic Analysis[e] - SA
4. INGESTION			
a. Milk	4 locations as follows: Samples from milking animals at 3 locations within 8 km distance having the highest dose potential (when available)[h] 1 sample from milking animals at a control location greater than 15 km distance from the site and in a less prevalent wind direction	With animals on pasture - SM At other times - M	Gamma isotopic analysis[e] and I-131 analysis - SM (when animals are on pasture); At other times - M
b. Fish and Invertebrates	4 locations as follows: 3 samples of commercially and recreationally important species in the vicinity of the plant discharge: one free swimming	When in season - SA	Gamma isotopic analysis[e] on edible portions - SA

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number Of Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION (Continued)			
b. Fish and Invertebrates (Continued)	species; one bottom feeding species; and one shellfish species. 1 sample of a similarly edible species from an area not influenced by plant discharge to serve as a control sample.		
c. Broadleaf Vegetation	3 locations as follows: Samples of broadleaf vegetation grown in 2 sectors of historically higher D/Q values at the SITE BOUNDARY if milk sampling is not performed. 1 sample of a similar broadleaf vegetation grown at a distance of greater than 15 km from the site in a less prevalent wind direction if milk sampling is not performed.	When available - M	Gamma isotopic analysis[e] and I-131 analysis - M (when available)

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATION

(a) Specific parameters of distance and direction sector from the site, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12.1-1 in a table and figure(s) in the ODCM. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment, and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program. In lieu of a License Event Report and pursuant to Specification 6.9.1.8, identify the cause of the unavailability of samples for that pathway 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).

(b) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information with minimal fading.

(c) The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

(d) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12.1-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

(e) ... isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(f) The "upstream" sample shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. "Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.

(g) A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. 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 compositing period (e.g., monthly) in order to assure obtaining a representative sample.

(h) When less than three (3) milking animals are available for testing within an 8-km distance, sampling of broadleaf vegetation shall be performed as indicated in Table 3.12.1-1, 4.c, in lieu of milk sampling.

TABLE 3.12.1-2 (Continued)

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)	Broadleaf Vegetation (pCi/kg, wet)
H-3	3×10^4	-	-	-	-
Mn-54	1×10^3	-	3×10^4	-	-
Fe-59	4×10^2	-	1×10^4	-	-
Co-58	4×10^2	-	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 4.12.1-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Broadleaf Vegetation (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01	-	-	-	-
H-3	3000	-	-	-	-	-
Mn-54	15	-	130	-	-	-
Fe-59	30	-	260	-	-	-
Co-58, 60	15	-	130	-	-	-
Zn-65	30	-	260	-	-	-
Zr-Nb-95	15	-	-	-	-	-
I-131	1[c]	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	-	-

TABLE 4.12.1-1 (Continued)

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

TABLE NOTATION

(a) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

(b) The lower limit of detectability (LLD) is the smallest concentration of a radioactive material in an unknown sample that will be detected with a 95% probability with a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\sigma_b}{E \ V \ 2.22 \ Y \ \exp(-\lambda_i t_e)}$$

Where:

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

σ_b = $(N/t_b)^{1/2}$

= standard deviation of background (cpm)

N = background count rate (cpm)

t_b = time background counted for (min)

E = counting efficiency, as counts per disintegration

V = volume or mass of sample

2.22 = conversion factor (dpm/pCi)

Y = fractional radiochemical yield

λ_i = radioactive decay constant of ith nuclide (sec^{-1})

t_e = elapsed time between sample collection and counting (sec)

TABLE 4.12.1-1 (Continued)

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

Typical values of E, V, Y, and t_e should be used in the calculation. It should be recognized that the LLD is defined as an "a priori" (before the fact) limit representing the capability of a measurement system and not as an "a posteriori" (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.7.

(c) The LLD of gamma isotopic analysis may be used.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 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 3.12.1-1(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 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.8.

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 3.12.1, 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 6.9.1.8, 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.

4.12.2 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 6.9.1.7.

NOTE: See Bases 3/4.12.2

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

NOTE: See Bases 3/4.12.3

MONITORING INSTRUMENTATION (Continued)

3/4.3.5.6 CHLORIDE INTRUSION MONITORS

The chloride intrusion monitors provide adequate warning of any leakage in the condenser or hotwell so that actions can be taken to mitigate the consequences of such intrusion in the reactor coolant system. With only a minimum number of instruments available, increased sampling frequency provides adequate information for the same purpose.

3/4.3.5.7 FIRE DETECTION INSTRUMENTATION

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

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

3/4.3.5.8 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the 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 limiting condition for operation is being exceeded, takes the next appropriate action to comply with the specification.

3/4.3.5.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring 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

3/4.3.5.9 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (Continued)

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, the main condenser offgas treatment system monitor, and the explosive gas monitoring instrumentation shown in Table 3.3.5.9-1 are not considered effluent monitoring instrumentation in the same sense as the other instrumentation listed in the table. Therefore, their alarm/trip setpoints are not necessarily set to ensure that the limits of Specification 3.11.2.1 are not exceeded.

The main condenser air ejector monitoring instrumentation channels are provided to monitor and control gross radioactivity removed from the main condenser. The alarm/trip setpoints for the main condenser air ejector monitor are set to ensure that the limits of Specification 3.11.2.7 are not exceeded. 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.

This specification also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the offgas treatment system (hydrogen monitors). The hydrogen monitors will become applicable when the offgas recombiners become fully operational (prior to operation of the Augmented Off-Gas Treatment System) 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.

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

3/4.3.6 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within an envelope of study events given in General Electric Company Topical Report NEDO-10349, dated March, 1971.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is a safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive

3/4.3.6 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to one EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch for each of the other two turbine stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for closure of the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic operating bypass at < 30% of RATED THERMAL POWER are annunciated in the control room.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents 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 HASL 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 limiting condition for operation is being exceeded, takes the next appropriate action to comply with the specification.

Note that for batch releases, recirculation of at least two tank volumes shall be considered adequate for thorough mixing.

The stabilization pond and service water liquid release types represent potential release pathways and not actual release pathways. Surveillance of these pathways is intended to alert the plant to a potential problem; analysis for principal gamma emitters should be sufficient to meet this intent. If analysis for principal gamma emitters indicates a problem (i.e., exceeds the trigger level of $5x10^{-6}$ µCi/ml), then complete sampling and analyses shall be performed as per Table 4.11.1-2. The trigger level of $5x10^{-6}$µCi/ml was chosen as being sufficient to provide reasonable assurance of accountability of all nuclides released based upon lower limits of detection and expected concentrations.

3/4.11.1.2 DOSE - LIQUID EFFLUENTS

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

DOSES (Continued)

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 to UNRESTRICTED AREAS 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. This specification is written for a two unit site.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

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 10 CFR 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 two reactor units having a shared liquid radwaste treatment system.

3/4.11.1.4 LIQUID HOLDUP TANKS

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 the tank and its associated piping are all Seismic Class I.

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.

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

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

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 for UNRESTRICTED AREAS. 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 in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II, of 10 CFR Part 20 [10 CFR Part 20.106 (b)]. For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC 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 a MEMBER OF THE PUBLIC 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.

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).

DOSE RATE (Continued)

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

3/4.11.2.2 DOSE-NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and, at the same time, implement the guides set forth in Section IV.A of Appendix I, to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS 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. The limits of this specification are twice the 10 CFR 50 Appendix I per reactor guidelines because they are written for a two unit site.

3/4.11.2.3 DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

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 to UNRESTRICTED AREAS 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

DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM (Continued)

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. The limits of this specification are twice the 10 CFR 50 Appendix I per reactor guidelines because they are written for a two unit site.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

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.

Until such time as the Augmented Off-Gas Treatment System becomes operational at the Brunswick Steam Electric Plant, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute offgas holdup line and stack filter house filtration. After the Augmented Off-Gas Treatment System becomes operational, the GASEOUS RADWASTE TREATMENT SYSTEM shall refer to the 30-minute offgas holdup line, stack filter house filtration, and the Augmented Off-Gas Treatment System.

3/4.11.2.5 VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTILATION EXHAUST TREATMENT SYSTEM (Continued)

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. At the Brunswick Steam Electric Plant, the only VENTILATION EXHAUST TREATMENT SYSTEMS shall be those installed for the Turbine Buildings' ventilation.

3/4.11.2.6 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas 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 Off-Gas Recombiners become fully operational (prior to operation of the Augmented Off-Gas Treatment System) 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.

3/4.11.2.7 MAIN CONDENSER AIR EJECTOR RADIOACTIVITY RELEASE RATE

Restricting the release rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at or beyond the exclusion boundary 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. 243,600 microcuries/second is equal to 100 microcuries/second/MWt for a rated thermal power of 2,436 MWt.

3/4.11.2.8 DRYWELL VENTING OR PURGING

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.

3/4.11.3 SOLID RADIOACTIVE WASTE

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.

TOTAL DOSE (40 CFR PART 190) (Continued)

3/4.11.4 TOTAL DOSE (40 CFR PART 190)

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 generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. 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 fulfills 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 Specification 3/4.11. 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.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

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 effluent 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 analyses are tabulated in terms of the Lower Limits of Detection (LLDs). The LLDs required by Table 4.12.1-1 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 Determination 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.

3/4.12.2 LAND USE CENSUS

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.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

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.

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 CONTAINMENT

CONFIGURATION

5.2.1 The PRIMARY CONTAINMENT is a steel-lined, reinforced concrete structure composed of a series of vertical right cylinders and truncated cones which form a drywell. This drywell is attached to a suppression chamber through a series of vents. The suppression chamber is a concrete, steel-lined pressure vessel in the shape of a torus. The primary containment has a minimum free air volume of 288,000 cubic feet.

DESIGN TEMPERATURE AND PRESSURE

5.2.2 The primary containment is designed and shall be maintained for:

a. Maximum internal pressure 62 psig.

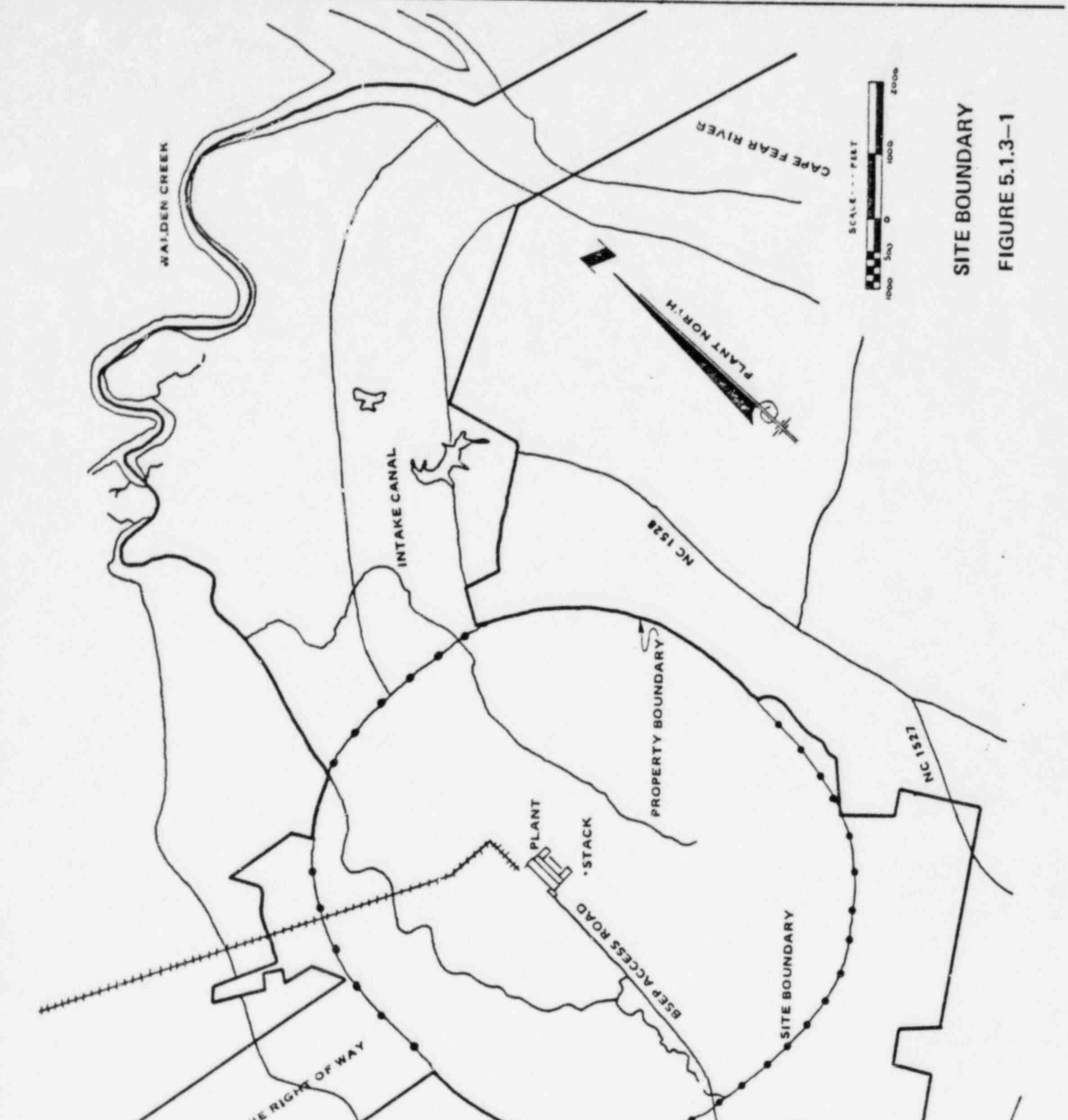
b. Maximum internal temperature: drywell 300°F
Suppression chamber 200°F

c. Maximum external pressure 2 psig.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 560 fuel assemblies with each 7 x 7 fuel assembly containing 49 fuel rods, each 8 x 8 fuel assembly containing 63 fuel rods; and each 8 x 8R fuel assembly containing 62 fuel rods. All fuel rods shall be clad with Zircaloy 2. The nominal active fuel length of each fuel rod shall be 144 inches for 7 x 7 fuel assemblies, 146 inches for 8 x 8 fuel assemblies, and 150 inches for 8 x 8R fuel assemblies. Each fuel rod shall contain a maximum total weight of 4430 grams of UO_2.



SITE BOUNDARY

FIGURE 5.1.3–1

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant 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. The facility duty shift shall be composed of at least the minimum facility shift crew composition shown in Table 6.2.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor.

c. When either reactor is in CONDITION 1, 2, or 3, at least one licensed Senior Reactor Operator shall be in the control room

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.

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* The individual qualified to implement radiation protection procedures and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

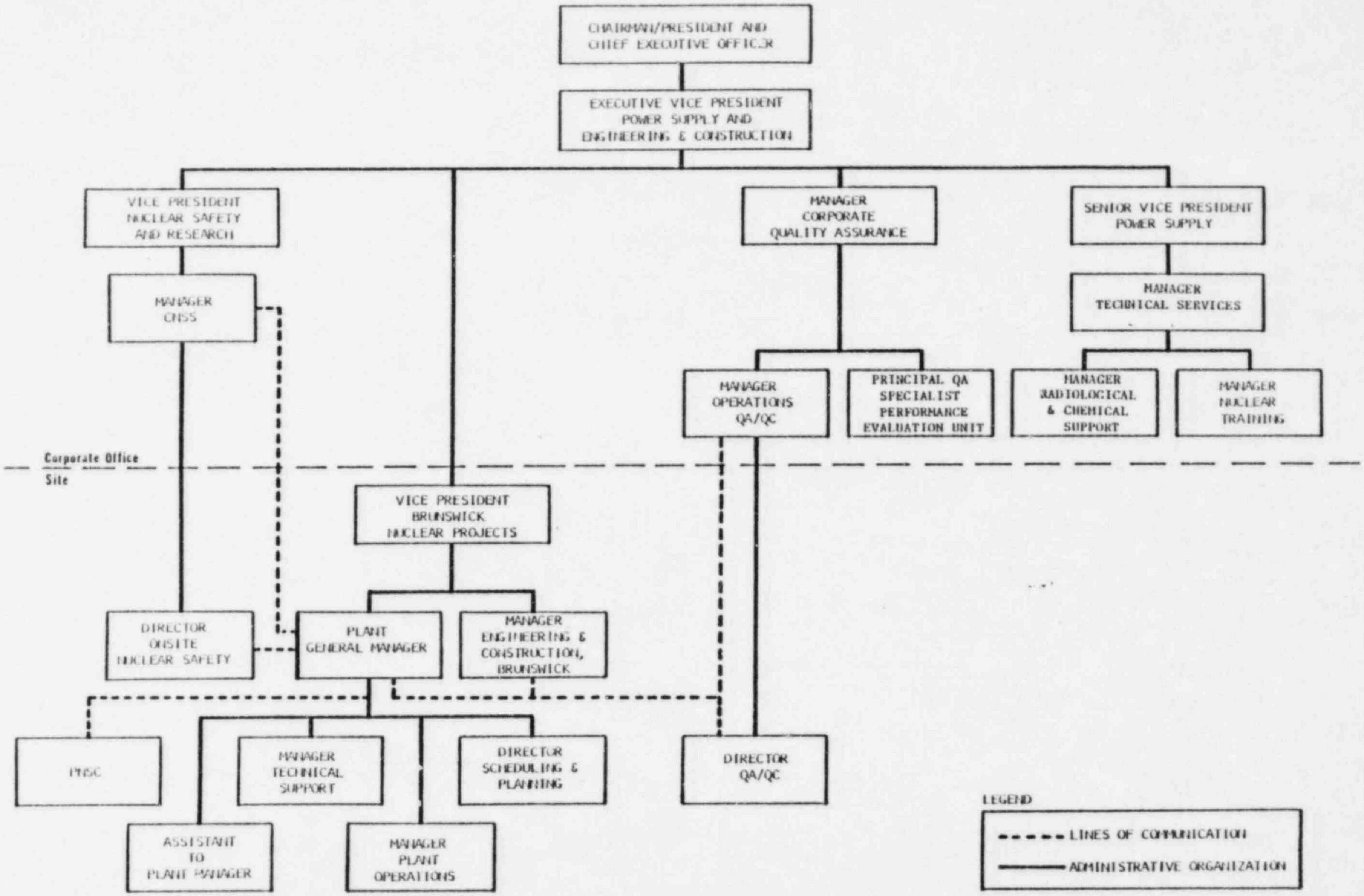
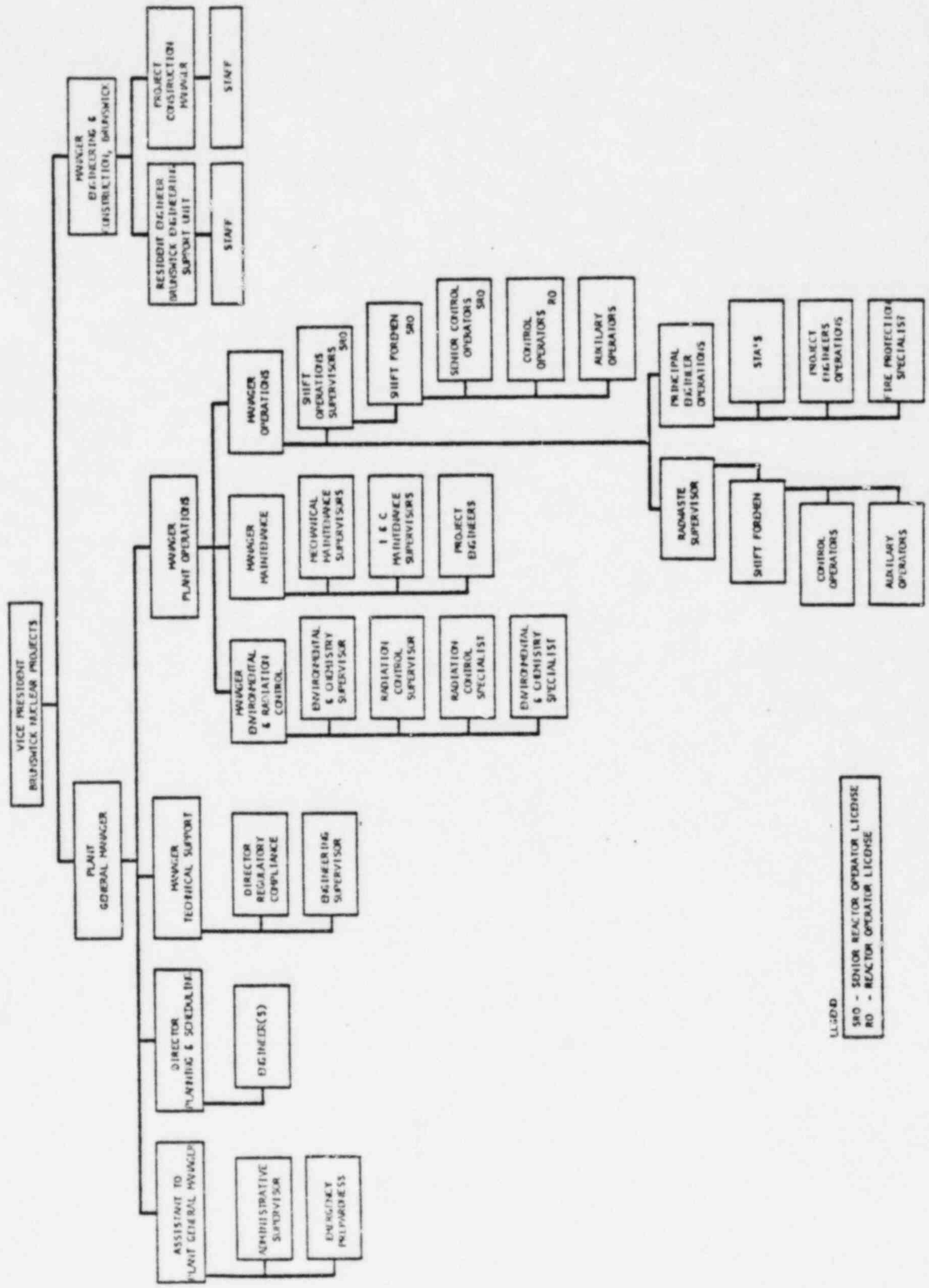

CHAIRMAN/PRESIDENT AND CHIEF EXECUTIVE OFFICER
EXECUTIVE VICE PRESIDENT POWER SUPPLY AND ENGINEERING & CONSTRUCTION
VICE PRESIDENT NUCLEAR SAFETY AND RESEARCH
MANAGER CORPORATE QUALITY ASSURANCE
SENIOR VICE PRESIDENT POWER SUPPLY
MANAGER CNSS
MANAGER TECHNICAL SERVICES
MANAGER OPERATIONS QA/QC
PRINCIPAL QA SPECIALIST PERFORMANCE EVALUATION UNIT
MANAGER RADIOLOGICAL & CHEMICAL SUPPORT
MANAGER NUCLEAR TRAINING
Corporate Office
Site
VICE PRESIDENT BRUNSWICK NUCLEAR PROJECTS
DIRECTOR ONSITE NUCLEAR SAFETY
PLANT GENERAL MANAGER
MANAGER ENGINEERING & CONSTRUCTION, BRUNSWICK
PNSC
MANAGER TECHNICAL SUPPORT
DIRECTOR SCHEDULING & PLANNING
DIRECTOR QA/QC
ASSISTANT TO PLANT MANAGER
MANAGER PLANT OPERATIONS
LEGEND
LINES OF COMMUNICATION
ADMINISTRATIVE ORGANIZATION

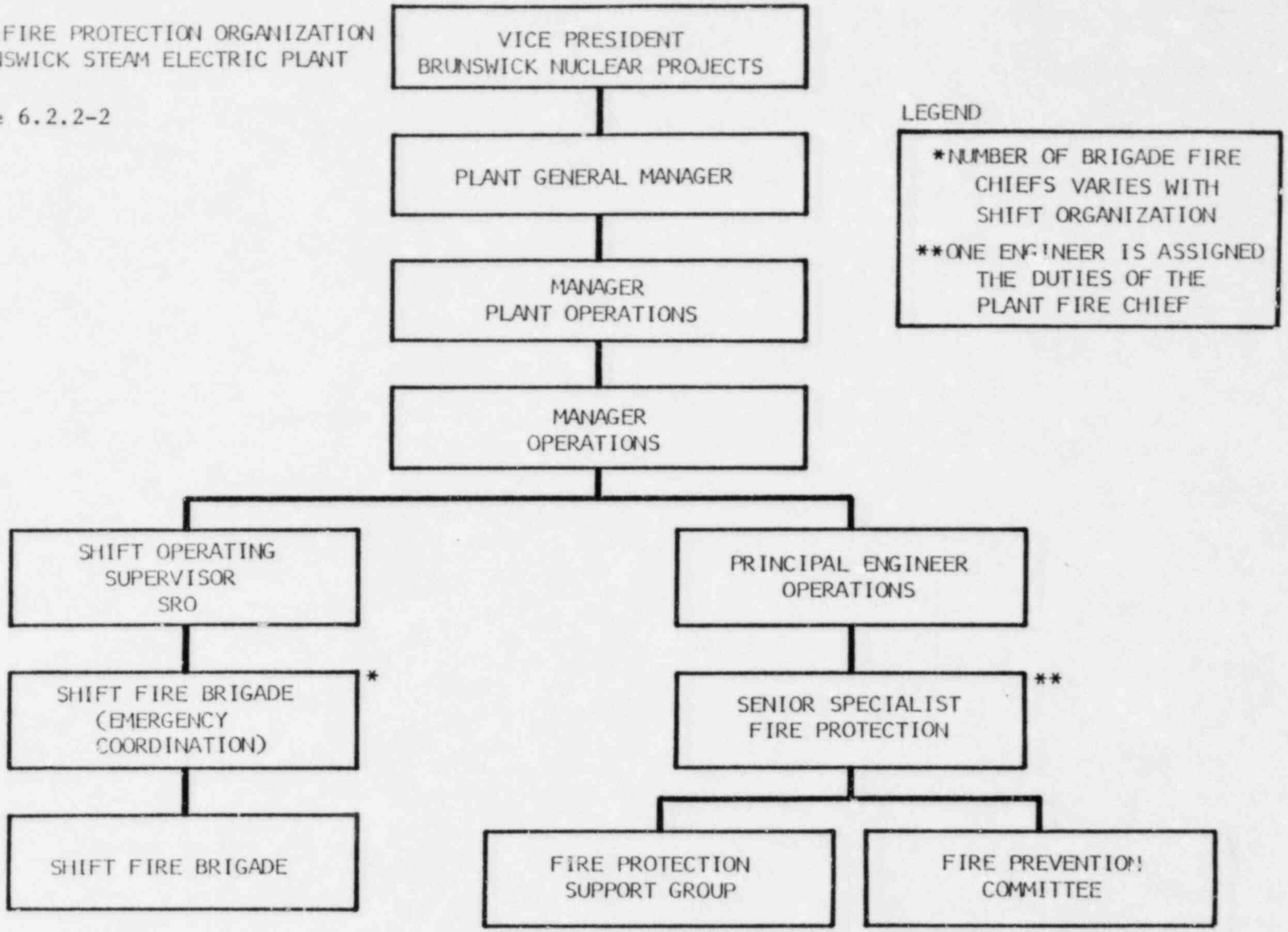
Figure 6.2.2-1

FACILITY ORGANIZATION



PLANT FIRE PROTECTION ORGANIZATION
BRUNSWICK STEAM ELECTRIC PLANT

Figure 6.2.2-2



6.2.3 ONSITE NUCLEAR SAFETY (ONS)

FUNCTION

6.2.3.1 The ONS Unit shall function to examine facility operating characterisitics, NRC issues, industry advisories, and other sources which may indicate areas for improving facility safety.

RESPONSIBILITIES

6.2.3.2 The ONS Unit shall be responsible for maintaining surveillance of facility activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.3 The ONS Unit shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving facility safety to the Manager-Corporate Nuclear Safety.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall serve in an advisory capacity to the Shift Operating Supervisor on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.3 FACILITY STAFF QUALIFICATION

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Manager-Environmental & Radiation Control who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

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* Not responsible for sign-off function.

TABLE 6.2.2-1
MINIMUM FACILITY SHIFT CREW COMPOSITION

WITH UNIT 1 IN CONDITION 1, 2, OR 3		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[b]
SRO[a]	1	1[b]
RO[a]	3	3
AO	3	3
STA	1	1

WITH UNIT 1 IN CONDITION 4 OR 5		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[b]
SRO[a]	1	1[b]
RO[a]	3	3
AO	3	3
STA	1	None

WITH UNIT 1 DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	CONDITIONS 1, 2, & 3	CONDITIONS 4 & 5
SOS	1	1[b]
SRO[a]	1	1[b]
RO[a]	2	2
AO	3	3
STA	1	None

TABLE 6.2.2-1 (Continued)

MINIMUM FACILITY SHIFT CREW COMPOSITION

TABLE NOTATION

SOS - Shift Operating Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Auxiliary Operator (non-licensed individual)
STA - Shift Technical Advisor

(a) Assumes each individual is licensed on both plants.

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

The Shift Crew Composition 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.

6.4 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 Manager-Operations and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975.

6.5 REVIEW AND AUDIT

6.5.1 NUCLEAR SAFETY REVIEWERS

6.5.1.1 Individuals shall be designated/approved by the Plant General Manager for the first party and second party safety reviews.

6.5.1.2 Individuals designated under 6.5.1.1 above shall have an academic degree in an engineering or related field or equivalent and two years related experience.

6.5.1.3 A list shall be maintained of individuals qualified to perform first party and second party safety reviews.

6.5.1.4 The list specified in 6.5.1.3 above shall include individuals, in addition to the first party and second party safety reviewers, whose expertise may be necessary during the reviews to assure that the reviewers collectively possess the background and qualfications in the disciplines necessary and important to the specific review.

6.5.1.5 The list specified in 6.5.1.3 and 6.5.1.4 above shall include the disciplines for which each individual is qualified.

6.5.2 SAFETY REVIEW AND CONTROL

PROCEDURES, TESTS, AND EXPERIMENTS

6.5.2.1 A safety evaluation for the procedures required by Specification 6.8, other procedures that affect nuclear safety, and changes thereto shall be prepared.

6.5.2.2 A safety evaluation for proposed tests and experiments that affect nuclear safety shall be prepared.

PROCEDURES, TESTS, AND EXPERIMENTS (Continued)

6.5.2.3 The safety evaluation prepared in accordance with 6.5.2.1 or 6.5.2.2 above shall include a written determination, with basis, of whether or not the procedures required by Specification 6.8, other procedures that affect nuclear safety, proposed tests and experiments, and changes thereto constitutes an unreviewed safety question as defined in 10 CFR 50.59, or involves a change to the Technical Specifications.

6.5.2.4 A first party safety review of the safety evaluation prepared in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 above shall be performed by an individual qualified under Section 6.5.1. The first party reviewer may be the preparer of the safety evaluation.

6.5.2.5 A second party safety review of the safety evaluation prepared and reviewed in accordance with 6.5.2.1, 6.5.2.2, 6.5.2.3, and 6.5.2.4 above shall be performed prior to approval and implementation of the proposed changes.

6.5.2.6 The second party safety review shall be performed by an individual qualified under Section 6.5.1. The second party safety reviewer shall be an individual other than the individual who was the original preparer or the first party safety reviewer.

6.5.2.7 Following the first party and second party safety review, the procedures required by Specification 6.8, other procedures that affect nuclear safety, proposed tests or experiments, and changes thereto (other than editorial or typographical) which have been determined to not involve an unreviewed safety question as defined in 10 CFR 50.59 or change to the Technical Specifications shall be approved prior to implementation by the Plant General Manager, the designated alternate to the Plant General Manager, or the Manager of the functional area affected by the procedure, proposed test or experiment. These procedures, proposed tests or experiments, and changes thereto shall be approved by an individual other than those who performed the required safety reviews.

6.5.2.8 The procedures required by Specification 6.8 and changes thereto, other procedures that affect nuclear safety and changes thereto, and proposed tests or experiments and changes thereto that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications shall be reviewed by the Plant Nuclear Safety Committee and submitted to the Commission for approval prior to implementation. Matters of this kind shall be referred to the Corporate Nuclear Safety Section by the Plant General Manager or by other functional organizational units within Carolina Power & Light Company for review prior to implementation.

6.5.2.9 The procedures required by Specification 6.8 and changes thereto, any other procedures that affect nuclear safety and changes thereto, and proposed tests or experiments that constitute a change to the safety analysis report shall be referred to the Corporate Nuclear Safety Section for review pursuant to Specification 6.5.4.9.

PROCEDURES, TESTS, AND EXPERIMENTS (Continued)

6.5.2.10 Temporary changes to procedures required by Specification 6.8, any other procedures that affect nuclear safety, and proposed tests or experiments may be approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator License on the unit affected, if such a change does not change the intent of the original procedure, proposed test or experiment. Temporary changes shall be documented and, within 21 days of receiving approval, be reviewed pursuant to 6.5.2.1, 6.5.2.2, 6.5.2.3, 6.5.2.4, 6.5.2.5, 6.5.2.6, and 6.5.2.7 above.

MODIFICATIONS

6.5.2.11 A safety evaluation for all proposed modifications to plant systems or equipment that affect nuclear safety shall be performed.

6.5.2.12 The safety evaluation prepared shall include a written determination, with basis, of whether or not the proposed modification is a change in the facility as described in the safety analysis report, involves a change to the Technical Specifications, or constitutes an unreviewed safety question as defined in 10 CFR 50.59.

6.5.2.13 A first party safety review of the safety evaluation prepared in accordance with 6.5.2.11 and 6.5.2.12 above shall be performed by an individual qualified under Section 6.5.1. The first party safety reviewer may be the preparer of the safety evaluation.

6.5.2.14 A second party safety review of the safety evaluation prepared and reviewed in accordance with 6.5.2.11, 6.5.2.12, and 6.5.2.13 above shall be performed prior to approval of the proposed modification.

6.5.2.15 The second party safety review shall be performed by an individual qualified under Section 6.5.1. The second party safety reviewer shall be an individual other than the individual who was the original preparer or the first party safety reviewer.

6.5.2.16 Following the first party and second party safety review, proposed modifications which have been determined to not involve an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications shall be approved prior to implementation by the Plant General Manager or the designated alternate of the Plant General Manager. The proposed modifications shall be approved by an individual other than those who performed the required safety reviews.

6.5.2.17 Proposed modifications that are determined to constitute an unreviewed safety question as defined in 10 CFR 50.59 or a change to the Technical Specifications shall be reviewed by the Plant Nuclear Safety Committee and submitted to the Commission for approval prior to implementation. Matters of this kind shall be referred to the Corporate Nuclear Safety Section for review prior to implementation.

MODIFICATIONS (Continued)

6.5.2.18 Proposed modifications that constitute a change to the safety analysis report shall be referred to the Corporate Nuclear Safety Section for review pursuant to Specification 6.5.4.9.

OPERATING LICENSE CHANGES

6.5.2.19 A safety evaluation for all proposed changes to the Technical Specifications and the Operating License shall be prepared.

6.5.2.20 The safety evaluation prepared shall include a written preliminary determination, with basis, of whether or not the proposed Technical Specification/Operating License change(s) is a change in the facility as described in the safety analysis report.

6.5.2.21 A first party safety review of the safety evaluation prepared in accordance with 6.5.2.19 and 6.5.2.20 above shall be performed by an individual qualified under Section 6.5.1. The first party safety reviewer may be the preparer of the safety evaluation.

6.5.2.22 A second party safety review of the safety evaluation prepared and reviewed in accordance with 6.5.2.19, 6.5.2.20, and 6.5.2.21 above shall be performed prior to approval of the proposed change(s) by the Plant Nuclear Safety Committee.

6.5.2.23 The second party safety review shall be performed by an individual qualfied under Section 6.5.1. The second party safety reviewer shall be an individual other than the individual who was the original preparer or the first party safety reviewer.

6.5.3 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.3.1 As an effective means for the regular review, overview, evaluation, and maintenance of plant operational safety, a Plant Nuclear Safety Committee (PNSC) shall be established.

6.5.3.2 The PNSC shall function through the utilization of subcommittees, audits, investigations, reports, and/or performance of reviews as a group.

COMPOSITION

6.5.3.3 The PNSC shall be composed of the:

Chairman:	Plant General Manager*
Member:	Manager - Plant Operations
Member:	Manager - Technical Support
Member:	Manager - Operations
Member:	Manager - Maintenance
Member:	Manager - Environmental & Radiation Control
Member:	Assistant to Plant General Manager
Member:	Director - QA/QC
Member:	Director - Regulatory Compliance

ALTERNATES

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

6.5.3.5 All alternates, shall as a minimum, meet the equivalent qualification criteria as specified for professional-technical personnel in Section 4.4 of ANSI N18.1-1971.

MEETING FREQUENCY

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

QUORUM

6.5.3.7 The minimum quorum of the PNSC necessary for the performance of the PNSC activities of the Technical Specifications shall consist of the PNSC Chairman or his designated alternate and three members including alternates.

ACTIVITIES

6.5.3.8 The PNSC activities shall include the following:

a. Review of all procedures required by Specification 6.8 and changes thereto and any other procedures and changes thereto, any of which constitute an unreviewed safety question or involve a change to the Technical Specifications.

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* Or designated alternate.

ACTIVITIES (Continued)

b. Review of all proposed tests or experiments that constitute an unreviewed safety question or involve a change to the Technical Specifications.

c. Review of all proposed modifications that constitute an unreviewed safety question or involve a change to the Technical Specifications.

d. Review of all proposed changes to the Technical Specifications and Operating License.

e. Review of reports on violations of Technical Specifications.

f. Performance of special reviews, investigations (or analyses), and reports thereon as requested by the Manager - Corporate Nuclear Safety Section.

g. Review of events requiring 24 hours written notification to the Commission.

h. Review of facility operations to detect potential nuclear safety hazards.

i. Annual review of the Security Plan.

j. Annual review of the Emergency Plan.

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section.

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

AUTHORITY

6.5.3.9 The PNSC shall provide written notification within 24 hours to the Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section of disagreement between recommendations of the PNSC and the actions contemplated by the Plant General Manager; however, the course determined by the Plant General Manager shall be followed.

RECORDS

6.5.3.10 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the provisions of these Technical Specifications. Copies shall be provided to the Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section.

6.5.4 CORPORATE NUCLEAR SAFETY SECTION

FUNCTION

6.5.4.1 The Corporate Nuclear Safety Section (CNSS) of the Corporate Nuclear Safety & Research Department shall function to provide independent review of significant plant changes, tests, and procedures; verify that REPORTABLE OCCURRENCES are investigated in a timely manner and corrected in a manner that reduces the probability of recurrence of such events; and detect trends that may not be apparent to a day-to-day observer.

ORGANIZATION

6.5.4.2 The individuals assigned responsibility for independent reviews shall be specified in technical disciplines. These individuals shall collectively have the experience and competence required to review activities in the following areas:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. administrative controls

i. seismic and environmental

j. quality assurance practices

6.5.4.3 The Manager - Corporate Nuclear Safety Section shall have an academic degree in an engineering or related field and, in addition, shall have a minimum of ten years related experience, of which a minimum of five years shall be in the operation and/or design of nuclear power plants.

ORGANIZATION (Continued)

6.5.4.4 The independent safety review program reviewers shall have an academic degree in an engineering or related field or equivalent and, in addition, shall have a minimum of five years related experience.

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

6.5.4.6 At least three individuals, qualified as discussed in 6.5.4.4 above shall review each item submitted under the requirements of Section 6.5.4.9.

6.5.4.7 Independent safety reviews shall be performed by individuals not directly involved with the activity under review or responsible for the activity under review.

6.5.4.8 The Corporate Nuclear Safety Section independent safety review program shall be conducted in accordance with written, approved procedures.

REVIEW

6.5.4.9 The Corporate Nuclear Safety Section shall perform reviews of the following:

a. All procedures required by Specification 6.8 and changes thereto that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications.

b. All proposed tests or experiments that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications.

c. All proposed modifications that constitute an unreviewed safety question as defined in 10 CFR 50.59 or involve a change to the Technical Specifications.

d. All procedures required by Specification 6.8 and changes thereto and all proposed tests or experiments that constitute a change to the safety analysis report. This review may be performed after appropriate management approval; implementation may proceed prior to completion of the review.

e. All proposed modifications that constitute a change to the facility as described in the safety analysis report. This review may be performed after appropriate management approval; implementation may proceed prior to completion of the review.

REVIEW (Continued)

f. All proposed changes to the Technical Specifications and Operating License.

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

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

i. Events requiring 24 hour written notification to the Commission.

j. Reports and minutes of the PNSC.

k. Any other matter involving safe operation of the nuclear power plant that the Manager - Corporate Nuclear Safety Section deems appropriate for consideration or which is referred to the Manager - Corporate Nuclear Safety Section by the on-site operating organization or other functional organizational units within Carolina Power & Light Company.

6.5.4.10 Review of items considered under 6.5.4.9(g) through (i) above shall include the results of any investigations made and the recommendations resulting from these investigations to prevent or reduce the probability of recurrence of the event.

RECORDS

6.5.4.11 Records of Corporate Nuclear Safety Section reviews, including recommendations and concerns, shall be prepared and distributed as indicated below:

a. Copies of documented reviews shall be retained in the CNS files.

b. Recommendations and concerns shall be submitted to the Plant General Manager and Vice President - Brunswick Nuclear Projects, within 14 days of completion of the review.

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

6.5.5 CORPORATE QUALITY ASSURANCE AUDIT PROGRAM

FUNCTION

6.5.5.1 The Performance Evaluation Unit (PEU) of the Corporate Quality Assurance Department shall function to perform audits of facility activities specified in Specification 6.5.5.2.

AUDITS

6.5.5.2 Audits of facility activities shall be performed by the PEU. These audits shall encompass:

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

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

c. The results of actions taken to correct deficienies occurring in facility equipment, structures, systems, or methods 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 12 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. The radiological environmental monitoring program and the results thereof at least once per 12 months.

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

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

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

AUDITS (Continued)

l. Any other area of facility operation considered appropriate by the Corporate Quality Assurance Performance Evaluation Unit.

6.5.5.3 Personnel performing the quality assurance audits shall have access to the plant operating records.

RECORDS

6.5.5.4 Records of audits shall be prepared and retained.

6.5.5.5 Audit reports encompassed by 6.5.5.2 above shall be prepared, approved by the Principal QA Specialist - Performance Evaluation Unit, and forwarded to the Executive Vice President - Power Supply and Engineering and Construction; Vice President - Brunswick Nuclear Projects; Vice President - Corporate Nuclear Safety and Research; Plant General Manager; and others, as appropriate, within 30 days after completion of the audit.

AUTHORITY

6.5.5.6 The Principal QA Specialist - Performance Evaluation Unit under the Manager - Corporate Quality Assurance shall be responsible for the following:

a. The administering of the Corporate Quality Assurance Audit Program.

b. The approval of the individual(s) selected to conduct quality assurance audits.

PERSONNEL

6.5.5.7 Audit personnel shall be independent of the area audited.

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

6.5.5.9 Qualified outside consultants or other individuals independent from those personnel directly involved in plant operation shall be used to augment the audit teams when necessary.

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM

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

6.5.6 OUTSIDE AGENCY INSPECTION AND AUDIT PROGRAM (Continued)

6.5.6.2 An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 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 Plant General Manager and submitted to the Manager - Corporate Nuclear Safety and the Vice President - Brunswick Nuclear Projects.

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 NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President - Brunswick Nuclear Projects and the Manager - Corporate Nuclear Safety Section shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant General Manager. 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 Vice President - Brunswick Nuclear Projects, and the Manager - Corporate Nuclear Safety Section within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

SAFETY LIMIT VIOLATION (Continued)

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. OFFSITE DOSE CALCULATION MANUAL implementation.

h. PROCESS CONTROL PROGRAM implementation.

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

6.9 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 Regional Administrator, United States Nuclear Regulatory Commission, Region II, unless otherwise noted.

STARTUP REPORTS

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

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

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all

STARTUP REPORTS (Continued)

three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation, supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS1/

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

PERSONNEL EXPOSURE AND MONITORING REPORT2/

6.9.1.5 Reports required on an annual basis shall include a tabulation of the number of station, utility, and other personnel, including contractors, receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions2/, e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated, based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources shall be assigned to specific major work functions.

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

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3/]

6.9.1.6 Routine radiological environmental operating reports covering the operation of the facility during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.1.7 The Annual Radiological Environmental Operating Report shall include the following:

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

b. Results of land use censuses required Specification 3.12.2.

c. Results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the OFFSITE DOSE CALCULATION MANUAL, as well as summarized and tabulated results of these analyses and measurements in the format of Table 3 in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data should be submitted as soon as possible in a supplementary report.

d. A summary description of the radiological environmental monitoring program.

e. At least two legible maps[4/] covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor.

f. The results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

g. Discussion of all deviaions from the sampling schedule of Table 3.12.1-1.

h. Discussion of all analyses in which the LLD required by Table 4.12.1-1 was not achievable.

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

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[5/]

6.9.1.8 Routine radioactive effluent release reports covering the operation of the facility during the previous 6 months of operation shall be submitted within the time periods specified in Specifications 6.9.1.9 and 6.9.1.10 below.

6.9.1.9 The portion of the Semiannual Radioactive Effluent Release Reports to be submitted within 60 days after January 1 and July 1 of each year shall include the following:

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

b. The information specified below for each class of solid waste (as defined by 10 CFR Part 61, when implemented) shipped offsite during the report period:

   1. Container volume,

   2. Total curie quantity (specified whether determined by measurement or estimate).

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

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

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

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

c. A list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioctive materials in gaseous and liquid effluents made during the reporting period.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[5/]

6.9.1.8 Routine radioactive effluent release reports covering the operation of the facility during the previous 6 months of operation shall be submitted within the time periods specified in Specifications 6.9.1.9 and 6.9.1.10 below.

6.9.1.9 The portion of the Semiannual Radioactive Effluent Release Reports to be submitted within 60 days after January 1 and July 1 of each year shall include the following:

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

b. The information specified below for each class of solid waste (as defined by 10 CFR Part 61, when implemented) shipped offsite during the report period:

   1. Container volume,

   2. Total curie quantity (specified whether determined by measurement or estimate),

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

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

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

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

c. A list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioctive materials in gaseous and liquid effluents made during the reporting period.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

d. Any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) or 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 3.12.2.

6.9.1.10 The portion of the Semiannual Radioactive Effluent Release Report to be submitted within 90 days after January 1 of each year shall include the following:

a. An annual summary of hourly meteorological data collected over the previous calendar year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.6/

b. An assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the station during the previous calendar year.

MONTHLY OPERATING REPORTS

6.9.1.11 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system safety/relief valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the U. S. Nuclear Regulatory Commission, Region II, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.12 The REPORTABLE OCCURRENCES of Specifications 6.9.1.13 and 6.9.1.14 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

6.9.1.13 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region II,

6/ In lieu of submission with the Semiannual 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.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of the 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.

   Note: Instrument drift discovered as a result of testing need not be reported under this item (but see 6.9.1.13.e, 6.9.1.13.f, and 6.9.1.14.a below).

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.

   Note: If specified action is taken when a system is found to be operating between the most conservative and least conservative aspects of a limiting condition for operation listed in the Technical Specifications, the limiting condition for operation is not considered to have been violated and no report need be submitted under this section (but see 6.9.1.14.b below).

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

   Note: Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this section.

d. Reactivity anomalies involving disagreement with 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.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

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

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

Note: For 6.9.1.13.e and 6.9.1.13.f, reduced redundancy that does not result in loss of system function need not be reported under this section (but see 6.9.1.14.b and 6.9.1.14.c below).

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.

Note: This item is intended to provide for reporting of potentially generic problems.

THIRTY DAY WRITTEN REPORTS

6.9.1.14 The types of events listed below shall be the subject of written reports to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region II within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of the 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.

THIRTY DAY WRITTEN REPORTS (Continued)

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 (but see 6.9.1.13.a and 6.9.1.13.b above).

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 (but see 6.9.1.13.b above).

Note: Routine surveillance testing, instrument calibration, or preventive maintenance that require configurations described in 6.9.1.14.a and 6.9.1.14.b above need not be reported except where test results themselves reveal a degraded mode as described above.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems (but see 6.9.1.13.f above).

d. Abnormal degradation of systems other than those specified in 6.9.1.13.c above designed to contain radioactive material resulting from the fission process.

Note: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, U. S. Nuclear Regulatory Commission, Region II within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.5.1.

b. Seismic event analysis, Specification 4.3.5.1.2.

c. Reactor coolant specific activity analysis, Specification 3.4.5.

d. Fire detection instrumentation, Specification 3.3.5.7.

e. Fire suppression systems, Specifications 3.7.7.1, 3.7.7.2, 3.7.7.3, and 3.7.7.5.

SPECIAL REPORTS (Continued)

f. ECCS actuation, Specifications 3.5.3.1 and 3.5.3.2.

g. Fire barrier penetation, Specification 3.7.8.

h. Liquid Effluents Dose, Specification 3.11.1.2.

i. Liquid Radwaste Treatment, Specification 3.11.1.3

j. Dose - Noble Gases, Specification 3.11.2.2.

k. Dose - Iodine-131, Tritium, and Radionuclides in Particulate Form, Specification 3.11.2.3.

l. Gaseous Radwaste Treatment, Specification 3.11.2.4.

m. Ventilation Exhaust Treatment, Specification 3.11.2.5.

n. Total Dose, Specification 3.11.4.

o. Monitoring Program, Specification 3.12.1.b.

6.10 RECORD RETENTION

Facility records shall be retained in accordance wtih 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 OCCURRENCES 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 detector 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 inservice 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 (1) meetings of the PNSC, (2) meetings of the previous off-site review organization, the Company Nuclear Safety Committee (CNSC), and (3) the independent reviews performed by the Corporate Nuclear Safety Section.

m. Records for Environmental Qualification which are covered under the provisions of paragraph 6.16.

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

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "Control Device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 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 (RWP)*. 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 Radiation 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 Operations Shift Foreman on duty and/or the qualified plant health physicist.

* Health Physics personnel or personnel escorted by 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 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.13.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the ODCM:

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

 1. 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);

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

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

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

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 The PROCESS CONTROL PROGRAM (PCP) shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the PCP:

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

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

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

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

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

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS 7/

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

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

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

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

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

4. An evaulation of the change that shows the predicted release of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change 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 amendments thereto

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

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

8. Documentation of the fact that the change was reviewed and found acceptable to the PNSC.

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

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

6.16 ENVIRONMENTAL QUALIFICATION

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

ENCLOSURE 3

JUSTIFICATIONS FOR MAJOR VARIANCES BETWEEN
THE BRUNSWICK RETS SUBMITTAL AND
NUREG-0472/0473 GUIDANCE

VARIANCE NO. 1: Many types of instrumentation are included in the proposed RETS submittal which have conditional applicabilities (i.e., instrumentation applicability is dependent on future modification and installation). This instrumentation includes: stabilization pond discharge instrumentation, reactor building component cooling (service water) monitors, Augmented Off-Gas (AOG) precooler service water monitor, salt water release tanks, main condenser offgas hydrogen monitors, and the main condenser offgas treatment system (AOG) noble gas monitor.

REFERENCE: Proposed Technical Specifications 3/4.3.5.8, 3/4.3.5.9, and 3/4.11.1.1.

JUSTIFICATION FOR VARIANCE: All of the above instrumentation is included in the enclosed RETS submittal because they will have a direct bearing on Brunswick's capability to monitor its effluents. However, none of this instrumentation is currently installed and operational. Over the next few years plans are to have all of this instrumentation engineered, procured, and installed. Some of this instrumentation will probably be made operational before the RETS are implemented. The submittal has been written with conditional applicabilities such that the referenced Technical Specifications can apply to the current situation at Brunswick and will continue to apply after each of the above instrumentation becomes operational. Therefore, we can minimize the number of Technical Specification changes which would be required over the next few years.

VARIANCE NO. 2: In satisfying guidance for inclusion of component cooling water monitors in the liquid instrumentation section of the proposed RETS submittal, the reactor building's component cooling (service water) monitors to be installed in the future are included, but component cooling water monitors for the turbine buildings are not included.

REFERENCE: Proposed Technical Specification 3/4.3.5.8.

JUSTIFICATION FOR VARIANCE: At Brunswick, the main service water monitor is downstream of all the component cooling water systems and, therefore, provides monitoring for these potential release pathways. However, in order to provide early detection of very small leaks from components of the reactor building component cooling water system, we plan to install monitors on the reactor building component cooling water system in the future and are including the monitors in the enclosed RETS submittal. No monitors for the turbine building component cooling water system have been included because the service water (effluent) lines of that system are at a greater pressure than the closed loop lines of the cooling system. Any leaks which may develop would be into the closed loop lines and not into the service water lines.

VARIANCE NO. 3: The source check frequency for the liquid radwaste effluent monitor is "at least once per 31 days" in the proposed RETS submittal rather than NUREG-0472's "prior to each release."

REFERENCE: Proposed Technical Specification 3/4.3.5.8.

JUSTIFICATION FOR VARIANCE: Performing a source check on the liquid radwaste effluent monitor prior to each release is impractical at Brunswick due to the large number of releases (nominally approximately 25/week) and the time required to perform an adequate source check (approximately two hours). The monitor does not have a "built-in" check source. In order to perform a source check, the detector has to be physically removed from its process location and introduced to a radioactive source. In addition to the time that would be lost from processing liquid effluents, there are ALARA concerns over employees accumulating unnecessary dose by performing frequent source checks. Each release of liquid radwaste serves as a process source check in that monitor response to the radioactive effluent (source) is verified. This verification, along with a monthly source check using a known source, meets the intent of more frequent monitor source checks.

VARIANCE NO. 4: The proposed RETS submittal does not include of the following instrumentation requested by NUREG-0473:

- Auxiliary Building Ventilation Monitoring
- Fuel Storage Area Ventilation Monitoring
- Radwaste Area Ventilation Monitoring
- Turbine Gland Seal Condenser Vent and Mechanical Vacuum Pump Exhaust Monitoring

REFERENCE: Proposed Technical Specification 3/4.3.5.9.

JUSTIFICATION FOR VARIANCE: All of these systems are substreams of other systems included in the enclosed RETS submittal. Ventilation exhausts from the radwaste area, turbine gland seal condenser vent, and the mechanical vacuum pumps are directed via the main stack. The fuel storage area is inside the Reactor Building. There is no Auxiliary Building at the Brunswick Plant.

VARIANCE NO. 5: The proposed RETS submittal states that, upon experiencing an inoperable noble gas activity monitor on the Reactor Building Ventilation Monitoring System, releases can continue via this pathway provided that grab samples are collected at least once per 12 hours (Action 123) while NUREG-0473 guidance would require that releases via this pathway be suspended (Action 124).

REFERENCE: Proposed Technical Specification 3/4.3.5.9.

JUSTIFICATION FOR VARIANCE: Action 124 is intended for reactor building noble gas monitors which provide purge monitoring and can automatically isolate secondary containment. At Brunswick, drywell purges are routed to the main stack. The reactor building's roof vent noble gas monitor included in the submittal cannot initiate secondary containment isolation. This function is performed by a separate monitor which is upstream of the roof vent monitor in the ventilation system's ductwork. Technical Specifications currently exist in Appendix A for this separate monitor (see Specification 3/4.3.2). The reactor building roof vent noble gas monitor serves a function which is no different than that of the noble gas monitors on the turbine building vents or main stack. Therefore, Action 123 should correspond to the noble gas activity monitor for the Reactor Building Ventilation Monitoring System included in the enclosed RETS submittal.

VARIANCE NO. 6: The instrumentation requirements for the main condenser offgas treatment system explosive gas monitoring system (hydrogen monitors) have been somewhat reworded from NUREG-0472/0473 guidance due to the nature of the system at Brunswick.

REFERENCE: Proposed Technical Specification 3/4.3.5.9.

JUSTIFICATION FOR VARIANCE: The Brunswick gaseous waste treatment system will have two recombiner trains with two hydrogen monitors each. Each train is capable of handling 100% of the load. Nominally, both trains will be operated at 50% capacity. The enclosed RETS submittal has been written so that it is clear that there are two recombiner trains with two hydrogen monitors on each train.

VARIANCE NO. 7: Stabilization pond and service water liquid release types as described in the Radioactive Liquid Waste Sampling and Analysis Program are considered to be potential release pathways and as such are analyzed for principal gamma emitters to provide a mechanism to initiate an analysis program consistent with NUREG-0473 guidance should these potential release pathways become actual release pathways.

REFERENCE: Proposed Technical Specification 3/4.11.1.1.

JUSTIFICATION FOR VARIANCE: If analysis for principal gamma emitters indicates that an actual release pathway exists, then complete analyses shall be performed. The trigger level to initiate complete liquid analyses is based on having any gamma emitting radionuclide exceeding 5×10^{-6} μCi/ml in the sample. At Brunswick, the concentrations of most principal gamma emitters in liquid radwaste effluents are 100 times the concentrations of Sr-89 and Sr-90. The trigger level of 5×10^{-6} μCi/ml for principal gamma emitters would ensure initiation of a complete analysis program at Sr-89 and Sr-90 concentrations of 5×10^{-8} μCi/ml, the Lower Limit of Detection defined for Sr-89 and Sr-90.

VARIANCE NO. 8: Tritium analysis prior to each drywell purge as per NUREG-0473's guidance is not included in the proposed RETS submittal's Radioactive Waste Sampling and Analysis Program.

REFERENCE: Proposed Technical Specification 3/4.11.2.1.

JUSTIFICATION FOR VARIANCE: Gaseous tritium analyses are performed at the environmental release points on a monthly basis. From these samples estimates are made of the tritium released in gaseous effluents for the month. The gaseous tritium released as a result of drywell purges is insignificant compared to the total gaseous tritium release estimate.

The following example demonstrates a conservative estimate of tritium in the drywell under normal operating conditions:

- Volume of drywell = 4.65×10^9 cc = 1.64×10^5 ft^3
- Normal drywell temperature = 130°F
- (lbs. of H_2O vapor/lbs. of air) saturated at 130°F = 0.112
- At 130°F air is 17.52 ft^3/lb.

- Therefore, the total weight of the drywell air is:
  $(1.64 \times 10^5 \text{ft}^3)/(17.52 \text{ ft}^3/\text{lb}) = 9.36 \times 10^3$ lbs.
- With 0.112 lbs of water vapor/lb of air there are:
  $(9.36 \times 10^3)\ (0.112) = 1.04 \times 10^3$ lbs. of H_2O

  In volume, this would be:
  $[(1.04 \times 10^3 \text{ lb.})/8.4 \text{ lbs./gal})]\ (3.78 \text{ liter/gal})\ (1000 \text{ ml/liter})$

  $= 4.71 \times 10^5$ ml of H_2O

Assume that the total water vapor content in the drywell air is due to reactor water.

Nominal tritium concentration in reactor water = 5×10^{-4} µCi/ml

Therefore, the total activity in the drywell atmosphere would be:
$(4.71 \times 10^5 \text{ ml}) (5 \times 10^{-4} \mu\text{Ci/ml}) = 2.36 \times 10^2 \mu\text{Ci}$

Therefore, the concentration of tritium in the drywell atmosphere would be:
$(2.36 \times 10^2 \mu\text{Ci})/(4.65 \times 10 \text{ ml}) = 5.07 \times 10^{-8} \text{ Ci/ml}$.

This is much less than maximum permissible concentrations for gaseous tritium in unrestricted areas (2×10^{-7} µCi/ml according to 10 CFR 20, Appendix B, Table II, column 1).

During the first six months of 1982, twelve drywell purges were performed. Assuming that 2.36×10^2 µCi of tritium (as calculated above were released with each purge, then 2.83×10^{-3} Ci of tritium were released via the stack as a result of drywell purges. The total estimated tritium released via the stack for this six month period is 8.23×10^{-1} Ci. Therefore, tritium released as a result of drywell purges represents 0.34% of the total tritium released via the stack.

From this analysis, it is obvious that tritium analysis of the drywell prior to each purge is unnecessary.

VARIANCE NO. 9: Weekly tritium sampling of the ventilation from the spent fuel pool area is only required in the proposed RETS submittal if a "trigger level" of tritium concentration in the spent fuel pool water is exceeded. Instead of NUREG-0472's notation: "Tritium grab samples shall be taken at least once per seven days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool," the proposed RETS submittal includes a note which reads: "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 seven 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 seven days during refueling."

REFERENCE: Proposed Technical Specification 3/4.11.2.1.

JUSTIFICATION FOR VARIANCE: Gaseous tritium analysis at the Reactor Building ventilation exhaust is currently performed on a monthly basis. This frequency is deemed to be adequate. Only during refueling would there be a higher probability of tritium in the spent fuel pool than at other times. Fuel pool water would be the source of airborne tritium in the Reactor Building atmosphere. We prefer to sample the source rather than the atmosphere. During refueling, the fuel pool water will be sampled and analyzed for tritium at least once per seven days. If the tritium concentration in the water exceeds 2×10^{-4} µCi/ml , then sampling and analysis for gaseous tritium in the Reactor Building ventilation exhaust will be performed at least once per seven days. 2×10^{-4} µCi/ml tritium concentration in fuel pool water was chosen somewhat arbitrarily as an action level; however, we feel it is an acceptable action level in light of the fact that nominal tritium concentration in reactor water is 5×10^{-4} µCi/ml and the maximum permissible concentration for tritium in water in unrestricted areas is 3×10^{-3} µCi/ml as per 10 CFR 20, Appendix B, Table II, column 2.

VARIANCE NO. 10: The proposed RETS submittal does not require drywell venting or purging to take place through the Standby Gas Treatment (SBGT) System as opposed to NUREG-0473's guidance.

REFERENCE: Proposed Technical Specification 3/4.11.2.8.

JUSTIFICATION FOR VARIANCE: .&L prefers not to be required to always use the SBGT System for routine purging of the drywell. The SBGT System's primary purpose is for post-LOCA emergency use. Use of the SBGT System for routine purges would decrease the system's filter efficiency and could render the system unavailable for emergency use (i.e., if a LOCA were to occur while the SBGT System was being used to purge the drywell, then the filters might be destroyed by a pressure spike and be unavailable for use during recovery from the LOCA). Note: Sample analysis of the containment atmosphere is performed prior to each purge, and purging is performed via the main stack (a monitored pathway); therefore, releases from drywell purging operations should not result in exceeding the annual dose limits of 10 CFR Part 20 for unrestricted areas.

VARIANCE NO. 11: NUREG-0472/0473 guidance suggests that the monthly composite sample collection of surface water required by the Radiological Environmental Monitoring Program be collected with equipment that is capable of collecting an aliquot at hourly time intervals. Brunswick's proposed RETS submittal requires that the composite sample be collected with equipment that is capable of collecting an aliquot at 6 hour time intervals.

REFERENCE: Proposed Technical Specification 3/4.12.1.

JUSTIFICATION FOR VARIANCE: Brunswick's existing environmental composite sampling equipment does not have the capacity to make a monthly composite with aliquots being taken every hour. It does have the capacity to make a monthly composite with aliquots being taken once per 6 hours. It is deemed that aliquots that are taken at time intervals that are short (e.g., once per 6 hours) relative to the compositing period (e.g., monthly) is sufficient to assure obtaining a representative sample.

VARIANCE NO. 12: 10 CFR Part 50 requires that routine Radioactive Effluent Release Reports covering the operation of the plant during the previous 6 months of operation be submitted within 60 days after January 1 and July 1 of each year; in addition to the items normally included in these reports, NUREG guidance suggests that the report following January 1 of each year should also include an annual summary of meteorological data and an assessment of the radiation doses due to the plant's effluents during the previous calendar year. Brunswick's proposed RETS submittal allows that this additional information be due within 90 days after January 1 of each year rather than NUREG-0472's guidance of 60 days.

REFERENCE: Proposed Technical Specification 6.9.1.13.

JUSTIFICATION FOR VARIANCE: The current Appendix "B" Technical Specificiations for Brunswick require the preparation and submittal of all portions of the semi-annual report within 60 days following each report period. It has always been difficult for Brunswick to prepare and submit the portion of semi-annual report concerning dose and meteorology within 60 days of the end the semi-annual period. We believe that not enough time is allowed for adequate reviews of this information prior to submittal, thus increasing the likelihood of minor errors in the report. The major factor contributing to the difficulty of completing this portion of the report within 60 days has been the completion of Sr-89 and Sr-90 data analysis. Monthly composite samples are sent offsite for analysis, which usually takes four to five weeks to complete. Therefore, the strontium analysis for the last month in the reporting period and allowing some time for dose projections to be made, it can be seen that there is very little time left in the 60 days for adequate reviews. Allowing an extra 30 days in which the dose projections can be performed with adequate reviews will alleviate this problem.

VARIANCE NO. 13: NUREG-0472's guidance to include demonstration of conformance to 40 CFR Part 190 in the semiannual report due after January 1 of each year is omitted from the proposed RETS submittal.

REFERENCE: Proposed Technical Specification 6.9.1.13.

JUSTIFICATION FOR VARIANCE: Technical Specification 3.14.1 (for Total Dose) already provides a mechanism for demonstrating compliance with 40 CFR Part 190. This specification requires that an analysis be performed to demonstrate compliance with 40 CFR Part 190 only when doses due to effluents exceed twice any 10 CFR Part 20 or 10 CFR Part 50 limits and that this analysis be reported to the Commission by means of a Special Report. The intent of the specification obviously stems from NUREG-0543, Methods for Demonstrating Compliance With the EPA Uranium Fuel Cycle Standard (40 CFR Part 190), which states "As long as a nuclear plant site operates at a level below the Appendix I [10 CFR 50] reporting requirements, no extra analysis is required to demonstrate compliance with 40 CFR Part 190." For the Commission to require that compliance with 40 CFR Part 190 be demonstrated routinely in the semiannual report is to be inconsistent with the total dose specification and NUREG-0543. Since there is no regulation which requires that compliance with 40 CFR Part 190 to be demonstrated in a routine report, Brunswick prefers to omit this item from the semiannual report.

ENCLOSURE 4

BRUNSWICK STEAM ELECTRIC PLANT
OFFSITE DOSE CALCULATION MANUAL
(ODCM)