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DESIGN FEATURES

ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 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

1.4 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 instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

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

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

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

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

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

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1,

1.6.2 All equipment hatches are closed and sealed,

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ 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, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.11 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF 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

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.13 A GASEOUS RADWASTE TREATMENT SYSTEM is the 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. The system is composed of the waste gas decay tanks, regenerative heat exchanger, waste gas charcoal filters, process vent blowers, waste gas surge tanks and waste gas diaphram compressor.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are 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 leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.16 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and the specific monitoring locations of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, normal and emergency electrical 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 MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

1.21 The PROCESS CONTROL PROGRAM 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

1.22 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 confinement.

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE OCCURRENCE

1.26 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.8 and 6.9.1.9.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

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

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.31 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

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

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM is the 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

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

TABLE 1.1

OPERATIONAL MODES

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

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 1.2

FREQUENCY NOTATION

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

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, the unit shall be placed in a MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within 1 hour,
2. At least HOT SHUTDOWN within the next 6 hours, and
3. At least COLD SHUTDOWN within the following 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s), and device(s) are OPERABLE, or likewise satisfy the requirements of this Specification. Unless both conditions (1) and (2) are satisfied, the unit shall be placed in at least HOT STANDBY within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours.

This Specification is not applicable in MODES 5 or 6.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

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 or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-12. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Seminannual 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 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	26
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	1	26
b. Circulating Water System Effluent Line	1	29
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
a. Clarifier Effluent Line	1	26
4. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	27

TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
5. TANK LEVEL INDICATING DEVICES*		
a. Refueling Water Storage Tanks	1	28
b. Casing Cooling Storage Tanks	1	28
c. PG Water Storage Tanks**	1	28
d. Boron Recovery Test Tanks**	1	28

*Tanks included in this Specification are 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.

**This is a shared system with Unit 2.

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 26 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 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} microcurie/gram or an isotopic radioactivity at a lower limit of detection of at least $5x10^{-7}$ microcuries/gram.

ACTION 27 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Design capacity performance curves generated in situ may be used to estimate flow.

ACTION 28 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

ACTION 29 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, make repairs as soon as possible. Grab samples cannot be obtained via this pathway.

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	D	R	Q(1)
2. GROSS BETA OR GAMMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line	D	M	R	Q(2)
b. Circulating Water System Effluent Line	D	M	R	Q(2)
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR				
a. Clarifier Effluent Line	N.A.	N.A.	R	N.A.

TABLE 4.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
5. TANK LEVEL INDICATING DEVICES***				
a. Refueling Water Storage Tank	D*	N.A.	R	Q
b. Casing Cooling Storage Tank	D*	N.A.	R	Q
c. PG Water Storage Tanks**	D*	N.A.	R	Q
d. Boron Recovery Storage Tanks**	D*	N.A.	R	Q

*During liquid additions to the tank.
**This is a shared system with Unit 2.
***Tanks included in this Specification are 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.

TABLE 4.3-12 (Continued)

TABLE NOTATION

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

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

2. Instrument controls not set in operate mode.

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

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

2. Instrument controls not set in operate mode.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.11 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, 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 the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instruments to OPERABLE status within 30 days and, 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 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. PROCESS VENT SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	31,33
b. Iodine Sampler	1	*	31,34
c. Particulate Sampler	1	*	31,34
d. Process Vent Flow Rate Measuring Device	1	*	30
e. Sampler Flow Rate Measuring Device	1	*	30
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (Shared with Unit 2)			
a. Hydrogen Monitor	1	**	32
b. Oxygen Monitor	1	**	32

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER AIR EJECTOR SYSTEM			
a. Gross Activity Monitor	1	*	31
b. Flow Rate Monitor	1	*	30
4. VENTILATION VENT SYSTEM (Shared with Unit 2)			
a. Noble Gas Activity Monitor	1*	*	31
b. Iodine Sampler	1*	*	31
c. Particulate Sampler	1*	*	31
d. Flow Rate Monitor	1*	*	30
e. Sampler Flow Rate Monitor	1*	*	30

*One per vent stack.

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During process vent system operation (treatment for primary system offgases).

ACTION 30 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 31 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

ACTION 32 - With the number of channnels OPERABLE less than required by the minimum channels OPERABLE requirement, operation of this system may continue for up to 14 days provided grab samples are taken and analyzed daily. With this channel inoperable, operation may continue provided grab samples are taken and analyzed: (1) every 4 hours during degassing operations and (2) daily during other operations.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the Waste Gas Decay Tanks may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of Waste Gas Decay Tank effluents.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases from the Waste Gas Decay Tanks may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. PROCESS VENT SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	R	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Process Vent Flow Rate Measuring Device	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D(5)	N.A.	R	N.A.	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(3)	M	**
b. Oxygen Monitor	D	N.A.	Q(4)	M	**

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONDENSER AIR EJECTOR SYSTEM					
a. Noble Gas Activity Monitor	D	M	R	Q(2)	*
b. Flow Rate Monitor	D	N.A.	R	Q	*
4. VENTILATION VENT SYSTEM (Shared with Unit 2)					
a. Noble Gas Activity Monitor	D	M	R	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D(5)	N.A.	R	N.A.	*

TABLE NOTATION

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

** During process vent system operation (treatment for primary system offgases).

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

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

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

1. Instrument indicates measured levels above the alarm setpoint.
2. Instrument controls not set in operate mode.

(3) 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.

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

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

(5) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic or batch releases are made.

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) 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:

a. 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.

b. The provisions of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

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

4.11.1.1.2 The results of the 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.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

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

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

[c]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

[d]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.

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

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

[g]Whenever the secondary coolant activity exceeds 10^{-5} uCi/ml the turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to the maximum exposed MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of 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 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

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

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*The maximum exposed MEMBER OF THE PUBLIC and the critical organ is addressed in the ODCM.

LIQUID RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste ion exchanger 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 each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to the critical 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 sub-systems, 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

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

*The critical organ is addressed in the ODCM.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Refueling Water Storage Tank
b. Casing Cooling Storage Tank
c. PG Water Storage Tank*
d. Boron Recovery Test Tank*
e. Any Outside Temporary Tank**

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

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

*This is a shared system with Unit 2.

**Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste ion exchanger system.

RA IOACTIVE 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 5.1-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

b. The provisions of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 The dose rate due to 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 methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

*The critical organ is defined in the ODCM.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (uCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b]	$1x10^{-4}$
B. Containment PURGE	P Each PURGE Grab Sample	P Each PURGE	Principal Gamma Emitters[b]	$1x10^{-4}$
			H-3	$1x10^{-6}$
C. Process Vent Vent. Vent A Vent. Vent B	M[c,d,e] Grab Sample	M[c]	Principal Gamma Emitters[b]	$1x10^{-4}$
			H-3	$1x10^{-6}$
D. All Release Types as listed in A, B, C above.	Continuous[d]	W[e] Charcoal Sample	I-131	$1x10^{-12}$
	Continuous[d]	W[e] Particulate Sample	Principal Gamma Emitters[b]	$1x10^{-11}$
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	$1x10^{-11}$
	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr-90	$1x10^{-11}$
	Continuous[d]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-6}$

TABLE 4.11-2 (Continued)

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] (uCi/ml)
E. Condenser Air Ejector Vent[f]	W Grab Sample	W	Principle Gamma Emitters[g]	1×10^{-4}
Steam Generator Blowdown Vent[f]			H-3	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

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, Xe-135m, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD together with the above nuclides, shall also be identified and reported.

[c]Sampling and analysis shall also be performed following shutdown, startup, and whenever a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER occurs within a one hour period, if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is greater than 1.0 uCi/gm; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

[d]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.

[e]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one 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 applies if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is greater than 1.0 uCi/gm and; (2) the noble gas monitor shows that effluent activity has increased more than a factor of 3.

[f]Whenever the secondary coolant activity exceeds 10^{-5} uCi/ml, samples shall be obtained and analyzed weekly. The turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

[g]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, Xe-135m, and Xe-138 for gaseous emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD together with the above nuclides, shall also be identified and reported.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of 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 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.

RADIOACTIVE EFFLUENTS

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

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to the maximum exposed MEMBER OF THE PUBLIC from iodine-131, from tritium, and from all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to the critical organ* and,

b. During any calendar year: Less than or equal to 15 mrems to the critical 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.

*The critical organ is addressed in the ODCM.

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation over 31 days. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.3 mrem to the critical organ* over 31 days.

APPLICABILITY: At all times.

ACTION:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*The critical organ is addressed in the ODCM.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume and is less than 96% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas decay tanks greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas decay tanks greater than 4% volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas decay tanks shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas decay tanks with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to $\leq$ 25,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per month when the specific activity of the primary reactor coolant is $\leq$ 1.0 uCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 uCi/gm DOSE EQUIVALENT I-131, the Gas Storage Tank(s) shall be sampled once per 24 hours, when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 SOLIDIFICATION shall be conducted in accordance with a PROCESS CONTROL PROGRAM.

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 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to the maximum exposed 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 the critical 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 Specification 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 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.

SURVEILLANCE REQUIREMENTS

4.11.4 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.

*The critical organ is addressed in the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 4.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 4.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.11, 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 specified location exceeding the reporting levels of Table 4.12-2 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 actions 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 4.12-2 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 4.12-2 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. 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 vegetable samples unavailable from one or more of the sample locations required by Table 4.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12, identify the cause of the 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

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 4.12-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 4.12-1, the detection capabilities required by Table 4.12-3, and the guidance of the Radiological Assessment Branch Technical Position on Environmental Monitoring dated November, 1979, Revision No. 1.

TABLE 4.12-1

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]	36 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously to be placed as follows: 1) an inner ring of stations, one in each meteorological sector within the SITE BOUNDARY; an outer ring of stations, one in each meteorological sector within 8 km range from the site; the balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: a. 3 samples from close to the 3 SITE BOUNDARY locations (in different sectors) of the highest calculated historical annual average ground-level D/Q.	Continuous sampler (2/3 running time cycle), operation with sample collection weekly.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[c] Gamma isotopic analysis of composite (by

*The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sampling program.

TABLE 4.12-1 (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
	b. 1 sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q.		location) quarterly.[d]
	c. 1 sample from a control location 15-30 km distant and in the least prevalent wind direction.		
3. WATERBORNE			
a. Surface	a. 1 sample circulating water discharge	Sample off upstream, downstream and cooling lagoon. Grab Monthly.	Gamma isotopic analysis[d] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources only if likely to be affected.	Grab Quarterly	Gamma isotopic[d] and tritium analysis quarterly.
c. Sediment	1 sample from downstream area with existing or potential recreational value.	Semiannually	Gamma isotopic analysis[d] semiannually.

TABLE 4.12-1 (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			
a. Milk[f]	a. Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr[e].	Monthly at all times.	Gamma isotopic[d] and I-131 analysis monthly .
	b. 1 sample from milking animals at a control location (15-30 km distant and in the least prevalent wind direction).		
b. Fish and Invertebrates	a. 1 sample of commercially and recreationally important species (bass, sunfish, catfish) in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis on edible portions.
	b. 1 sample of same species in areas not influenced by plant discharge.		

TABLE 4.12-1 (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
c. Food Products (cont'd)	a. Samples of an ediable broad leaf vegetation grown nearest each of two different offsite locations of highest predicted historical annual average ground-level D/Q if milk sampling is not performed.	Monthly if available, or at harvest.	Gamma isotopic[d] and I-131 analysis.
	b. 1 sample of broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly if available, or at harvest.	Gamma isotopic[d] and I-131 analysis.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 4.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, "October 1978, and to Radiological Assessment Branch Technical Positions, Revision 1, November 1979. 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 6.9.11.1. 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 Licensee Event Report and pursuant to Specification 6.9.1.12, 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. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. 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.

TABLE 4.12-1 (Continued)

TABLE NOTATION

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[c]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.

[d]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[e]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[f]If milk sampling cannot be performed, use item 4.c.

TABLE 4.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

TABLE 4.12-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	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	10*	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		

*The LLD for gamma isotopic analysis shall be used.

TABLE 4.12-3 (Continued)

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 6.9.1.11.

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

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

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

Where:

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

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

E is the counting efficiency, as counts per disintegration,

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

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

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

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

TABLE 4.12-2 (Continued)

TABLE NOTATION

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. Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

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 6.9.1.12.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 25 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 control station location, having the lowest calculated dose or dose commitment(s), via the 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 6.9.1.12, identify the new location(s) 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).

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 results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

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*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.12-1.4c shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials (which contain nuclides produced at nuclear power stations) supplied as part of an Interlaboratory Comparison Program, described in the ODCM, 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 6.9.1.11.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be maintained as described in the ODCM.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

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

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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

3/4.11.1 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 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 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 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).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I 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 are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This Specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT

The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This Specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

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 that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This Specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC, either within or outside the SITE BOUNDARY to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC, who may at times be within the SITE BOUNDARY the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

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

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of 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).

3/4.11.2.2 DOSE - NOBLE GASES

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

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

This Secification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are 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. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate Specifications for iodine-131, radioiodines, tritium, and radionuclides 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 that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This Specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and 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 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.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this Specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

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.

3/4.11.4 TOTAL DOSE

This Specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 20 by 46 FR 18525. The Specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of ACTION that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and 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 Specifications 3.11.1 and 3.11.2. 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 the 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 and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The 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 are considered optimum for routine environmental measurements in industrial laboratories. 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.

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).

3/4.12.2 LAND USE CENSUS

This Specification is provided to ensure that changes in the use of areas 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 this census. The best information from the door-to-door survey, from aerial survey or from 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/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

3/4.12/3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved 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.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-1.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which allows identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

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

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 190 feet, 7 inches.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of concrete floor pad = 10 feet.

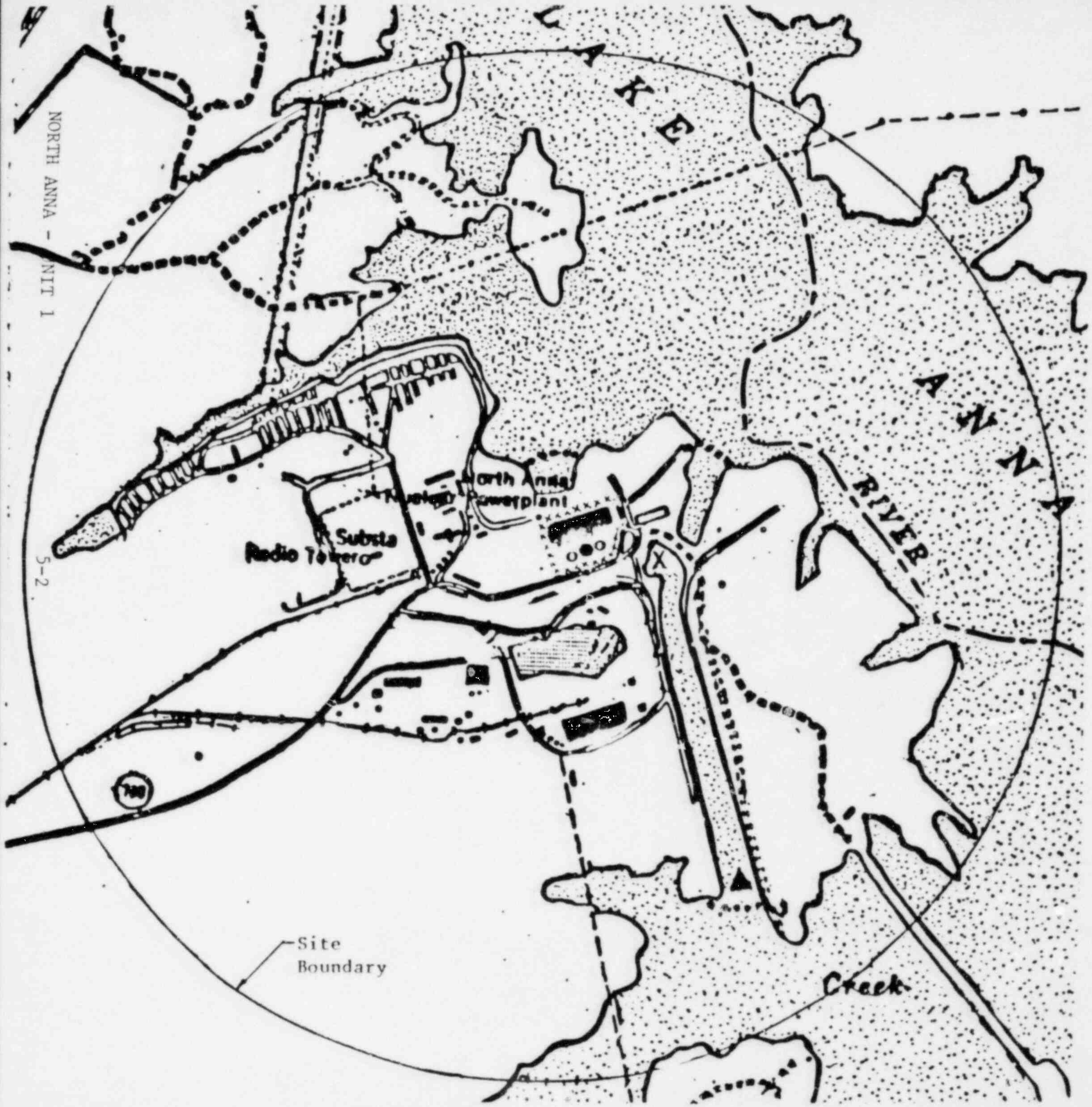
f. Nominal thickness of the cylindrial portion of the steel liner = 3/8 inches.

g. Net free volume = 1.825×10^6 cubic feet.

h. Nominal thickness of hemispherical dome portion of the steel liner = 1/2 inch.

DESIGN PRESSURE AND TEMPERATURE

5.2.2. The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.



● Gaseous Release
1. Process Vent- 157.5 ft.
2. Vent-Vent A&B and other release points considered ground level releases

X Liquid Release to the Discharge Canal

▲ Liquid Release to the Unrestricted Area

●●● Buoy Barriers

xxxx Security Fence- Area outside is unrestricted for gaseous effluents

Land Maximum Member of the Public Occupancy = 336 hrs/year

Lake Maximum Member of the Public Occupancy = 2232 hrs/year

Figure 5.1-1

Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents

MEETING FREQUENCY

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

QUORUM

6.5.1.5 A quorum of the SNSOC consists of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8.1 and changes thereto, 2) all programs required by Specification 6.8.4 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Station Manager to affect nuclear safety.

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

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

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

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

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

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

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

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

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

AUTHORITY

6.5.1.7 The SNSOC shall:

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

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

c. Provide written notification within 24 hours to Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control of disagreement between the SNSOC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

6.5.2 SAFETY EVALUATION AND CONTROL (SEC)

FUNCTION

6.5.2.1 SEC shall function to provide independent review of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Administrative controls and quality assurance practices

i. Other appropriate fields associated with the unique characteristics of the nuclear power plant

COMPOSITION

6.5.2.2 The SEC staff shall be composed of the Director-Safety Evaluation and Control and a minimum of three individuals who are qualified as staff specialists. Each SEC staff specialist shall have an academic degree in an engineering or physical science field and, in addition, shall have a minimum of five years technical experience in one or more areas given in Specification 6.5.2.1. These staff specialists shall not be directly involved in the licensing function.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the Director-Safety Evaluation and Control to provide expert advice to the SEC.

MEETING FREQUENCY

6.5.2.5 The SEC staff shall meet at least once per calendar month for the purpose of fostering interaction of reviews regarding safety-related operational activities.

REVIEW

6.5.2.7 The following subjects shall be reviewed by SEC:

a. Written safety evaluations of changes in the stations as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2) and is accomplished by review of minutes of the Station Nuclear Safety and Operating Committee and the design change program.

b. Proposed changes in procedures, proposed changes in the station, or proposed tests or experiments, any of which may involve a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2). Matters of this kind shall be referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee following its review prior to implementation.

REVIEW (Cont'd)

c. Changes in the Technical Specifications or license amendments relating to nuclear safety prior to implementation except in those cases where the change is identical to a previously reviewed proposed change.

d. Violations and reportable occurrences such as:

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

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

3. Reportable occurrences as defined in the station Technical Specification 6.9.1.8.

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

e. The Quality Assurance Department audit program at least once per 12 months and audit reports.

f. Any other matter involving safe operation of the nuclear power stations which a duly appointed subcommittee or committee member deems appropriate for consideration, or which is referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee.

g. Reports and meeting minutes of the Station Nuclear Safety and Operating Committee.

AUTHORITY

6.5.2.9 The Director-Safety Evaluation and Control shall report to and advise the Manager-Nuclear Technical Services, who shall advise the Vice President-Nuclear Operations on those areas of responsibility specified in Section 6.5.2.7.

RECORDS

6.5.2.10 Records of SEC activities required by Section 6.5.2.7 shall be prepared and maintained in the SEC files and a summary shall be disseminated as indicated below each calendar month.

1. Vice President-Nuclear Operations

2. Nuclear Power Station Managers

3. Manager-Nuclear Operations and Maintenance

4. Manager-Nuclear Technical Services
5. Executive Manager-Quality Assurance *
6. Others that the Director-Safety Evaluation and Control may designate.

6.5.3 QUALITY ASSURANCE DEPARTMENT

FUNCTION

6.5.3.1 The Quality Assurance Department shall function to audit station activities. These audits shall encompass:

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

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

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

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

e. The Station Emergency Plan and implementing procedures at least once per 12 months. *

f. The Station Security Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the Executive Manager-Quality Assurance or the Senior Vice President-Power Operations.

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

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

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

* Pending NRC approval.

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

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

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

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

AUTHORITY

6.5.3.2 The Quality Assurance Department shall report to and advise the Executive Manager-Quality Assurance, who shall advise the Senior Vice President-Power Operations on those areas of responsibility specified in Section 6.5.3.1.

RECORDS

6.5.3.3 Records of the Quality Assurance Department audits shall be prepared and maintained in the department files. Audit reports shall be disseminated as indicated below:

1. Nuclear Power Station Manager
2. Manager-Nuclear Operations and Maintenance
3. Manager-Nuclear Technical Services
4. Director-Safety Evaluation and Control *
5. Supervisor of area audited
6. Nuclear Power Station Manager Quality Assurance

* Pending NRC approval.

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 SNSOC and submitted to the Director-Safety Evaluation and Control and the Manager-Nuclear Operations and Maintenance.

6.7 SAFETY LIMIT VIOLATION

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

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

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager-Nuclear Operations and Maintenance, and the Director-Safety Evaluation and Control shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. 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 Director-Safety Evaluation and Control and the Manager-Nuclear Operations and Maintenance 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, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program Implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

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.

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

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

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

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

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements and

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

b. In-Plant Radiation Monitoring

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

(i) Training of personnel,

(ii) Procedures for monitoring, and

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

c. Secondary Water Chemistry

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

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

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

(iii) Identification of process sampling points,

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

(v) Procedures defining corrective actions for all control point chemistry conditions,

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

(vii) Monitoring of the condensate at the discharge of the condensate pumps for evidence of condenser inleakage. When condenser in-leakage is confirmed, the leak shall be repaired, plugged, or isolated within 96 hours.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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

STARTUP REPORTS

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

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

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

ANNUAL REPORTS[1/]

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

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b.).

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety 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 Regional Office, of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

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

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

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

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

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

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

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

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to copy with accidents analyzed in the SAR.

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

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

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

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

THIRTY-DAY WRITTEN REPORT

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of 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. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

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

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

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

CORE SURVEILLANCE REPORT

6.9.1.10 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) in all core planes containing Bank "D" control rods and in all unrodded core planes, the surveillance power level, P_m, for Technical Specifications 3.2.1 and 3.2.6, and the F_Q flyspeck basis shall be provided to the Director of the Regional Office of Inspection and Enforcement, with a copy to;

Director, Office of Nuclear Reactor Regulation
Attention: Chief of Core Performance Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

at least 60 days prior to cycle initial criticality. In the event that the limits would be submitted at some other time during core life, they shall be submitted 60 days prior to the date the limits would become effective unless otherwise exempted by the Commission.

Any additional information needed to support the F_{xy}^{RTP} AND P_m submittal will be by request from the NRC and need not be included in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

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

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

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position,

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

Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 4.12-1 and discussion of all analyses in which the LLD required by Table 4.12-3 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

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

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

The Radioactive Effluent Release Report shall be submitted within 60 days after January 1 of each year. This report shall include an assessment of the radiation doses to the maximum exposed MEMBERS OF THE PUBLIC due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. Annual meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This meteorological data shall be retained in a file on site and shall be made available to the NRC upon request. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the OFFSITE DOSE CALCULATION MANUAL (ODCM). Concurrent meteorological conditions or historical annual average atmospheric dispersion conditions shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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

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

If the dose to the maximum exposed MEMBER OF THE PUBLIC due to the radioactive liquid and gaseous effluents from the station during the previous calendar year exceeds twice the limits of Specification 3.11.1.2a, 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 the dose assessment shall include the contribution from direct radiation. The dose to the maximum exposed MEMBER OF THE PUBLIC shall show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The Radioactive Effluent Release Reports shall include a list of unplanned releases as required to be reported in Technical Specification 6.9.1.9.e from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirement of the applicable reference specification:

a. Inservice Inspection Program Reviews shall be reported within 90 days of completion. Specification 4.0.5

b. ECCS Actuation shall be reported within 90 days of the occurrence. The report shall describe the circumstances of the actuation and the total accumulated cycles to date. Specification 3.5.2 and 3.5.3.

c. With Seismic Monitoring Instrumentation inoperable for more than 30 days, submit a special report within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrumentation to operable status. Specification 3.3.3.3.

d. For all seismic events actuating a seismic monitoring instrument, submit a special report within 10 days describing the magnitude, frequency spectrum and resultant effects upon features important to safety. Specification 4.3.3.3.2.

e. With Meteorological Instrumentation inoperable for more than 7 days, submit a special report within the next 10 days, outlining the cause of the malfunction and the plans for restoring the instrumentation to operable status. Specification 3.3.3.4.

f. With the primary coolant specific activity > 1.0 uCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ uCi/gram, a specific activity analysis shall be included in the REPORTABLE OCCURRENCE report required pursuant to Specification 6.9.1.7. The information requested in Specification 3.4.8 shall also be included in that report.

g. With sealed source or fission detector leakage tests revealing the presence of $\geq$ 0.005 microcuries of removable contamination submit a special report on an annual basis outlining the corrective actions taken to prevent the spread of contamination. Specification 4.7.11.1.3.

h. With the MTC more positive than 0 Δ k/k/°F submit a special report within the next 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. Specification 3.1.1.4.

i. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10.1, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.1.

j. For any abnormal degradation of the containment structure detected during the performance of Specification 4.6.1.6.2, an initial report shall be submitted within 10 days after completion of Specification 4.6.1.6.2. A final report, which includes (1) a description of the condition of the liner plate and concrete, (2) inspection procedure, (3) the tolerance on cracking and (4) the corrective actions taken, shall be submitted within 90 days after the completion of Specification 4.6.1.6.2.

k. Inoperable Fire Detection Instrumentation, Specification 3.3.3.7.

l. Inoperable Fire Suppression Systems, Specifications 3.7.14.1, 3.7.14.2, 3.7.14.3, 3.7.14.4 and 3.7.14.5.

6.10 RECORD RETENTION

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

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

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

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

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

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

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

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

c. Records of facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components identified in Table 5.9-1.

g. Records of reactor tests and experiments.

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

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

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

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

l. Records of meetings of the SNSOC.

m. Records of meetings of the System Nuclear Safety and Operating Committee to issuance of Amendment No. 30.

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 would include procedures effective at specified times and QA records showing that these procedures were followed.

o. Records of secondary water sampling and water quality.

p. Records for Environmental Qualification which are covered under the provisions of Paragraph 6.13.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

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

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

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

c. An individual qualified in the 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.

*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 in high radiation areas.

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

6.13 ENVIRONMENTAL QUALIFICATION

6.13.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental

Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License NPF-4 dated October 24, 1980.

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

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Licensee initiated changes to the PCP:

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

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

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SNSOC.

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

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.15.2 Licensee initiated changes to the ODCM:

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

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

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SNSOC.

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

6.16 MAJOR CHANGES TO RADIOACTIVE SOLID WASTE TREATMENT SYSTEMS*

6.16.1 Licensee initiated major changes to the radioactive solid waste systems:

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

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

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

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

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

d. An evaluation of the change, in quantity of solid waste that differs from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change, which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

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

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

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

2. Shall become effective upon review and acceptance by SNSOC.

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2.0 LIMITING CONDITIONS FOR OPERATION

2.1 NON-RADIOLOGICAL - None

observations in a form consistent with National Weather Service procedures. Summaries of all data and observations shall be available to the NRC upon request.

Any modification to the onsite meteorological monitoring program as described above, or planned alterations of the area in the vicinity of the meteorological tower(s) that would interfere with the measurement of meteorological conditions representative of the site, will require written approval in accordance with Section 5.6.3.

Bases

The collection of meteorological data at the plant site will provide information which will be used to develop atmospheric diffusion parameters to estimate potential radiation doses to the public resulting from actual routine or abnormal releases of radioactive materials to the atmosphere, and to assess the actual impact of the plant cooling system on the atmospheric environment of the site area. A meteorological data collection program as described above is necessary to meet the requirements of subparagraph 50.36(a)(2) of 10 CFR Part 50, Appendix E to 10 CFR Part 50, and 10 CFR Part 51.

5.5.3 Procedures for Environmental Surveillance - Nonradiological

Not applicable.

5.5.4 Quality Assurance of Program Results

The procedures document shall provide for assurance of the quality of program results, including analytical measurements. This portion of the procedures document shall document the program in policy directives, designate a responsible organization or individuals, include purchased services (e.g., contractual lab or other contract services), include audits by licensee personnel, and include procedures for revising programs, systems to identify and correct deficiencies, investigate anomalous or suspect results, and review and evaluate program results and reports.

5.5.5 Changes in Procedures, Station Design or Operation

Changes in procedures, station design or operation may be made subject to conditions described below, provided such changes are approved by the SNSOC (Review and Audit responsibility per Section 5.3).

a. The licensee may (1) make changes in the station design and operation as described in the FES, FES Addendum and the Environmental Report, (2) make changes in the procedures described in the document developed in accordance with Subsection 5.5, and (3) conduct tests and experiments not described in the document developed in acordance with Subsection 5.5, without prior Commission approval, unless the proposed change, test or experiment involves a change in the objectives of the ETS, an unreviewed environmental question of substantive impact, or affects the requirements of Subsection 5.5.6 of these ETS.

b. A proposed change, test, or experiment shall be deemed to involve an unreviewed environmental question (1) if the probability of magnitude of environmental impact may be increased; or (2) if a possibility for a substantive environmental impact of a different type than any evaluated previously in the FES or FES Addendum may be created.

c. The licensee shall maintain records of changes in procedures and in facility design or operation made pursuant to this Subsection, to the extent that such changes constitute changes in procedures as described in the document developed in accordance with Section 5.5 or in the FES, FES Addendum and ER. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph "a" of this Subsection. These records shall include a written evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question of substantive impact or constitute a change in the objectives of these ETS, or affect the requirements of Section 5.5.6 of these ETS. The licensee shall furnish to the Commission, annually or at such shorter intervals as may be specified in the license, a report containing descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

d. Changes in the procedures developed in accordance with Subsection 5.5 which affect sampling frequency, location, gear, or replication shall be reported to the NRC within 30 days after their implementation. These reports shall describe the changes made, the reasons for making the changes, an evaluation of the environmental impact of these changes, and the statement required under the provisions of Subsection 5.5.6.

5.5.6 Consistency with Initially Approved Programs

Any modifications or changes of the procedures developed in accordance with Subsection 5.5 must be governed by the need to maintain consistency with previously used procedures so that direct comparisons of data are technically valid. Such modifications or changes must be justified and supported by adequate comparative sampling programs or studies demonstrating the comparability of results or which provide a basis for making adjustments that would permit direct comparisons.

These demonstrations of comparability shall be submitted to the NRC in accordance with the provision of Subsection 5.5.5 and 5.6.1 of these ETS.

5.6 Station Reporting Requirements

5.6.1 Routine Reports - None.

5.6.2 Nonroutine Reports

5.6.2.1 Nonroutine Non-Radiological Environmental Operating Report

None.

5.6.3 Changes in Environmental Technical Specifications

A report shall be made to the NRC prior to implementation of a change in plant design, in plant operation, or in procedures described in Section 5.5 if the change would have a significant effect on the environment or involves an environmental matter or question not previously reviewed and evaluated by the NRC. The report shall include a description and evaluation of the change and a supporting benefit-cost analysis.

Request for changes in Environmental Technical Specifications shall be submitted to the Director, Office of Nuclear Reactor Regulation, for review and authorization. The request shall include an evaluation of the environmental impact of the proposed changes and a supporting benefit-cost analysis.

5.6.4 Changes in Permits and Certifications

None

5.7 Records Retention

Records and logs relative to the following areas shall be made and retained for the life of the station:

a. Records and drawings detailing plant design changes and modifications made to systems and equipment as described in Section 5.6.3.

b. Reports from environmental monitoring, surveillance, and special surveillance and study activities required by these Environmental Technical Specifications.

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ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AXIAL FLUX DIFFERENCE

1.2 AXIAL FLUX DIFFERENCE shall be the difference in normalized flux signals, expressed in % of RATED THERMAL POWER between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.3 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

1.4 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 instrumentation channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.5 A CHANNEL FUNCTIONAL TEST shall be:

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

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

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

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

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

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1,

1.6.2 All equipment hatches are closed and sealed,

1.6.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.6.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

1.8 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\overline{E}$ - AVERAGE DISINTEGRATION ENERGY

1.10 $\overline{E}$ 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, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.11 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF 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

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

GASEOUS RADWASTE TREATMENT SYSTEM

1.13 A GASEOUS RADWASTE TREATMENT SYSTEM is the 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. The system is composed of the waste gas decay tanks, regenerative heat exchanger, waste gas charcoal filters, process vent blowers, waste gas surge tanks and waste gas diaphram compressor.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are 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 leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.16 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints and the specific monitoring locations of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, normal and emergency electrical 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 MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

1.21 The PROCESS CONTROL PROGRAM 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

1.22 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 confinement.

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.25 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE OCCURRENCE

1.26 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.8 and 6.9.1.9.

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.28 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

SOLIDIFICATION

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

SOURCE CHECK

1.30 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to radiation. This applies to installed radiation monitoring systems.

STAGGERED TEST BASIS

1.31 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

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

UNIDENTIFIED LEAKAGE

1.33 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM is the 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

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

TABLE 1.1

OPERATIONAL MODES

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

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

TABLE 1.2

FREQUENCY NOTATION

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

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

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 or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-12. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Seminannual 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 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-12.

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	26
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line	1	26
b. Circulating Water System Effluent Line	1	29
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR		
a. Clarifier Effluent Line	1	26
4. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	1	27

TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
5. TANK LEVEL INDICATING DEVICES*		
a. Refueling Water Storage Tanks	1	28
b. Casing Cooling Storage Tanks	1	28
c. PG Water Storage Tanks**	1	28
d. Boron Recovery Test Tanks**	1	28

*Tanks included in this Specification are 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.

**This is a shared system with Unit 1.

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 26 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 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} microcurie/gram or an isotopic radioactivity at a lower limit of detection of at least 5×10^{-7} microcuries/gram.

ACTION 27 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Design capacity performance curves generated in situ may be used to estimate flow.

ACTION 28 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

ACTION 29 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, make repairs as soon as possible. Grab samples cannot be obtained via this pathway.

TABLE 4.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	D	R	Q(1)
2. GROSS BETA OR GAMMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line	D	M	R	Q(2)
b. Circulating Water System Effluent Line	D	M	R	Q(2)
3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR				
a. Clarifier Effluent Line	N.A.	N.A.	R	N.A.

TABLE 4.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
5. TANK LEVEL INDICATING DEVICES***				
a. Refueling Water Storage Tank	D*	N.A.	R	Q
b. Casing Cooling Storage Tank	D*	N.A.	R	Q
c. PG Water Storage Tanks**	D*	N.A.	R	Q
d. Boron Recovery Storage Tanks**	D*	N.A.	R	Q

*During liquid additions to the tank.

**This is a shared system with Unit 1.

***Tanks included in this Specification are 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.

TABLE 4.3-12 (Continued)

TABLE NOTATION

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

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

2. Instrument controls not set in operate mode.

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

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

2. Instrument controls not set in operate mode.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.10.2.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the ODCM.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, 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 the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, for reasons other than a above, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instruments to OPERABLE status within 30 days and, 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 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.3-13.

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. PROCESS VENT SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	31,33
b. Iodine Sampler	1	*	31,34
c. Particulate Sampler	1	*	31,34
d. Process Vent Flow Rate Measuring Device	1	*	30
e. Sampler Flow Rate Measuring Device	1	*	30
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (Shared with Unit 1)			
a. Hydrogen Monitor	1	**	32
b. Oxygen Monitor	1	**	32

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER AIR EJECTOR SYSTEM			
a. Gross Activity Monitor	1	*	31
b. Flow Rate Monitor	1	*	30
4. VENTILATION VENT SYSTEM (Shared with Unit 1)			
a. Noble Gas Activity Monitor	1*	*	31
b. Iodine Sampler	1*	*	31
c. Particulate Sampler	1*	*	31
d. Flow Rate Monitor	1*	*	30
e. Sampler Flow Rate Monitor	1*	*	30

*One per vent stack.

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During process vent system operation (treatment for primary system offgases).

ACTION 30 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 31 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

ACTION 32 - With the number of channnels OPERABLE less than required by the minimum channels OPERABLE requirement, operation of this system may continue for up to 14 days provided grab samples are taken and analyzed daily. With this channel inoperable, operation may continue provided grab samples are taken and analyzed: (1) every 4 hours during degassing operations and (2) daily during other operations.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the Waste Gas Decay Tanks may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of Waste Gas Decay Tank effluents.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases from the Waste Gas Decay Tanks may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. PROCESS VENT SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	D	P	R	Q(1)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Process Vent Flow Rate Measuring Device	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D(5)	N.A.	R	N.A.	*
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(3)	M	**
b. Oxygen Monitor	D	N.A.	Q(4)	M	**

TABLE 4.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. CONDENSER AIR EJECTOR SYSTEM					
a. Noble Gas Activity Monitor	D	M	R	Q(2)	*
b. Flow Rate Monitor	D	N.A.	R	Q	*
4. VENTILATION VENT SYSTEM (Shared with Unit 1)					
a. Noble Gas Activity Monitor	D	M	R	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D(5)	N.A.	R	N.A.	*

TABLE 4.3-13 (Continued)

TABLE NOTATION

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

** During process vent system operation (treatment for primary system offgases).

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

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

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

1. Instrument indicates measured levels above the alarm setpoint.
2. Instrument controls not set in operate mode.

(3) 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.

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

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

(5) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic or batch releases are made.

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) 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:

a. 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.

b. The provisions of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

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

4.11.1.1.2 The results of the 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.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (mCi/ml)
A. Batch Releases[b,g]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P Each Batch	M Composite[d]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
B. Continuous Releases[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters[f]	$5x10^{-7}$
			I-131	$1x10^{-6}$
			Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	Continuous[f]	M Composite[f]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous[f]	Q Composite[f]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

TABLE 4.11-1 (Continued)

TABLE NOTATION

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

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

[c]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable at levels exceeding the LLD, together with the above nuclides, shall also be identified and reported.

[d]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.

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

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

[g]Whenever the secondary coolant activity exceeds 10^{-5} uCi/ml the turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to the maximum exposed MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of 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 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

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

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*The maximum exposed MEMBER OF THE PUBLIC and the critical organ is addressed in the ODCM.

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste ion exchanger 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 each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the total body or 0.2 mrem to the critical 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 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

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

*The critical organ is addressed in the ODCM.

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each of the following unprotected outdoor tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Refueling Water Storage Tank
b. Casing Cooling Storage Tank
c. PG Water Storage Tank*
d. Boron Recovery Test Tank*
e. Any Outside Temporary Tank**

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

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

*This is a shared system with Unit 1.

**Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste ion exchanger system.

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 5.1-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

b. The provisions of Specification 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 The dose rate due to 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 methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

*The critical organ is defined in the ODCM.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (uCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE	P Each PURGE Grab Sample	P Each PURGE	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. Process Vent Vent. Vent A Vent. Vent B	M[c] Grab Sample	M[c]	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as listed in A, B, C above.	Continuous[d]	W[e] Charcoal Sample	I-131	1×10^{-12}
	Continuous[d]	W[e] Particulate Sample	Principal Gamma Emitters[b]	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[d]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

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] (uCi/ml)
E. Condenser Air Ejector Vent[f]	W Grab Sample	W	Principle Gamma Emitters[g]	$1x10^{-4}$
Steam Generator Blowdown Vent[f]			H-3	$1x10^{-6}$

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

TABLE 4.11-2 (Continued)

TABLE NOTATION

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, Xe-135m, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD together with the above nuclides, shall also be identified and reported.

[c]Sampling and analysis shall also be performed following shutdown, startup, and whenever a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER occurs within a one hour period, if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is greater than 1.0 uCi/gm; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

[d]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.

[e]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one 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 applies if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant is greater than 1.0 uCi/gm and; (2) the noble gas monitor shows that effluent activity has increased more than a factor of 3.

[f]Whenever the secondary coolant activity exceeds 10^{-5} uCi/ml, samples shall be obtained and analyzed weekly. The turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours.

[g]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, Xe-135m, and Xe-138 for gaseous emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks that are measurable and identifiable, at levels exceeding the LLD together with the above nuclides, shall also be identified and reported.

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of 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 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.

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

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to the maximum exposed MEMBER OF THE PUBLIC from iodine-131, from tritium, and from all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to the critical organ* and,

b. During any calendar year: Less than or equal to 15 mrems to the critical 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.

*The critical organ is addressed in the ODCM.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases, from each reactor unit, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation over 31 days. The VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases, from each reactor unit, from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed 0.3 mrem to the critical organ* over 31 days.

APPLICABILITY: At all times.

ACTION:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*The critical organ is addressed in the ODCM.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume and is less than 96% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas decay tanks greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas decay tanks greater than 4% volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas decay tanks shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas decay tanks with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to $\leq$ 25,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per month when the specific activity of the primary reactor coolant is $\leq$ 1.0 uCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 uCi/gm DOSE EQUIVALENT I-131, the Gas Storage Tank(s) shall be sampled once per 24 hours, when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 SOLIDIFICATION shall be conducted in accordance with a PROCESS CONTROL PROGRAM.

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 6.9.1.9.b are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to the maximum exposed 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 the critical 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 Specification 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 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.

SURVEILLANCE REQUIREMENTS

4.11.4 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.

*The critical organ is addressed in the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 4.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 4.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.11, 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 specified location exceeding the reporting levels of Table 4.12-2 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 actions 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 4.12-2 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 4.12-2 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. 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 vegetable samples unavailable from one or more of the sample locations required by Table 4.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12, identify the cause of the 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

4.12.1.1 The radiological environmental monitoring samples shall be collected pursuant to Table 4.12-1 from the specific locations given in the table and figure(s) in the ODCM and shall be analyzed pursuant to the requirements of Table 4.12-1, the detection capabilities required by Table 4.12-3, and the guidance of the Radiological Assessment Branch Technical Position on Environmental Monitoring dated November, 1979, Revision No. 1.

TABLE 4.12-1

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]	36 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously to be placed as follows: 1) an inner ring of stations, one in each meteorological sector within the SITE BOUNDARY; an outer ring of stations, one in each meteorological sector within 8 km range from the site; the balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations.	Quarterly	Gamma dose quarterly.
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: a. 3 samples from close to the 3 SITE BOUNDARY locations (in different sectors) of the highest calculated historical annual average ground-level D/Q.	Continuous sampler (2/3 running time cycle), operation with sample collection weekly.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[c] Gamma isotopic analysis of composite (by

*The number, media, frequency, and location of samples may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an individual's dose and should be included in the sampling program.

TABLE 4.12-1 (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
	b. 1 sample from the vicinity of a community having the highest calculated annual average groundlevel D/Q.		location) quarterly.[d]
	c. 1 sample from a control location 15-30 km distant and in the least prevalent wind direction.		
3. WATERBORNE			
a. Surface	a. 1 sample circulating water discharge	Sample off upstream, downstream and cooling lagoon. Grab Monthly.	Gamma isotopic analysis[d] monthly. Composite for tritium analysis quarterly.
b. Ground	Samples from 1 or 2 sources only if likely to be affected.	Grab Quarterly	Gamma isotopic[d] and tritium analysis quarterly.
c. Sediment	1 sample from downstream area with existing or potential recreational value. ..	Semiannually	Gamma isotopic analysis[d] semiannually.

TABLE 4.12-1 (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			
a. Milk[f]	a. Samples from milking animals in 3 locations within 5 km distance having the highest dose potential. If there are none, then, 1 sample from milking animals in each of 3 areas between 5 to 8 km distant where doses are calculated to be greater than 1 mrem per yr[e].	Monthly at all times.	Gamma isotopic[d] and I-131 analysis monthly .
	b. 1 sample from milking animals at a control location (15-30 km distant and in the least prevalent wind direction).		
b. Fish and Invertebrates	a. 1 sample of commercially and recreationally important species (bass, sunfish, catfish) in vicinity of plant discharge area.	Semiannually.	Gamma isotopic analysis on edible portions.
	b. 1 sample of same species in areas not influenced by plant discharge.		

TABLE 4.12-1 (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
c. Food Products (cont'd)	a. Samples of an ediable broad leaf vegetation grown nearest each of two different offsite locations of highest predicted historical annual average ground-level D/Q if milk sampling is not performed.	Monthly if available, or at harvest.	Gamma isotopic[d] and I-131 analysis.
	b. 1 sample of broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed.	Monthly if available, or at harvest.	Gamma isotopic[d] and I-131 analysis.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[a] Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each and every sample location in Table 4.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, "October 1978, and to Radiological Assessment Branch Technical Positions, Revision 1, November 1979. 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 6.9.11.1. 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 Licensee Event Report and pursuant to Specification 6.9.1.12, 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. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. 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.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[c]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.

[d]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[e]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

[f]If milk sampling cannot be performed, use item 4.c.

TABLE 4.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	30,000				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

TABLE 4.12-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	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	10*	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		

*The LLD for gamma isotopic analysis shall be used.

TABLE 4.12-3 (Continued)

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 6.9.1.11.

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

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

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

Where:

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

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

E is the counting efficiency, as counts per disintegration,

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

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

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

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

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

TABLE 4.12-2 (Continued)

TABLE NOTATION

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. Analysis shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

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 residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

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 6.9.1.12.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 25 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 control station location, having the lowest calculated dose or dose commitment(s), via the 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 6.9.1.12, identify the new location(s) 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).

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 results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.12-1.4c shall be followed, including analysis of control samples.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials (which contain nuclides produced at nuclear power stations) supplied as part of an Interlaboratory Comparison Program, described in the ODCM, 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 6.9.1.11.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be maintained as described in the ODCM.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

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

3/4.3.3.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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

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 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 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 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).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I 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 are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This Specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT

The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This Specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

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 that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This Specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC, either within or outside the SITE BOUNDARY to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC, who may at times be within the SITE BOUNDARY the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

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

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of 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).

3/4.11.2.2 DOSE - NOBLE GASES

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

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

This Secification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are 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. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate Specifications for iodine-131, radioiodines, tritium, and radionuclides 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 that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This Specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and 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 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.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This Specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

The tanks included in this Specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification to a quantity that is less than the quantity which provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem in an event of 2 hours.

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Branch Technical Position ETSB 11-5 in NUREG-0800, July 1981.

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.

3/4.11.4 TOTAL DOSE

This Specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 20 by 46 FR 18525. The Specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of ACTION that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and 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 Specifications 3.11.1 and 3.11.2. 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 representative 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 the 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 and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-3 are considered optimum for routine environmental measurements in industrial laboratories. 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.

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).

3/4.12.2 LAND USE CENSUS

This Specification is provided to ensure that changes in the use of areas 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 this census. The best information from the door-to-door survey, from aerial survey or from 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/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

3/4.12/3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved 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.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-1.

MAP DEFINING UNRESTRICTED AREAS FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which allows identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

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

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 190 feet, 7 inches.

c. Minimum thickness of concrete walls = 4.5 feet.

d. Minimum thickness of concrete roof = 2.5 feet.

e. Minimum thickness of concrete floor pad = 10 feet.

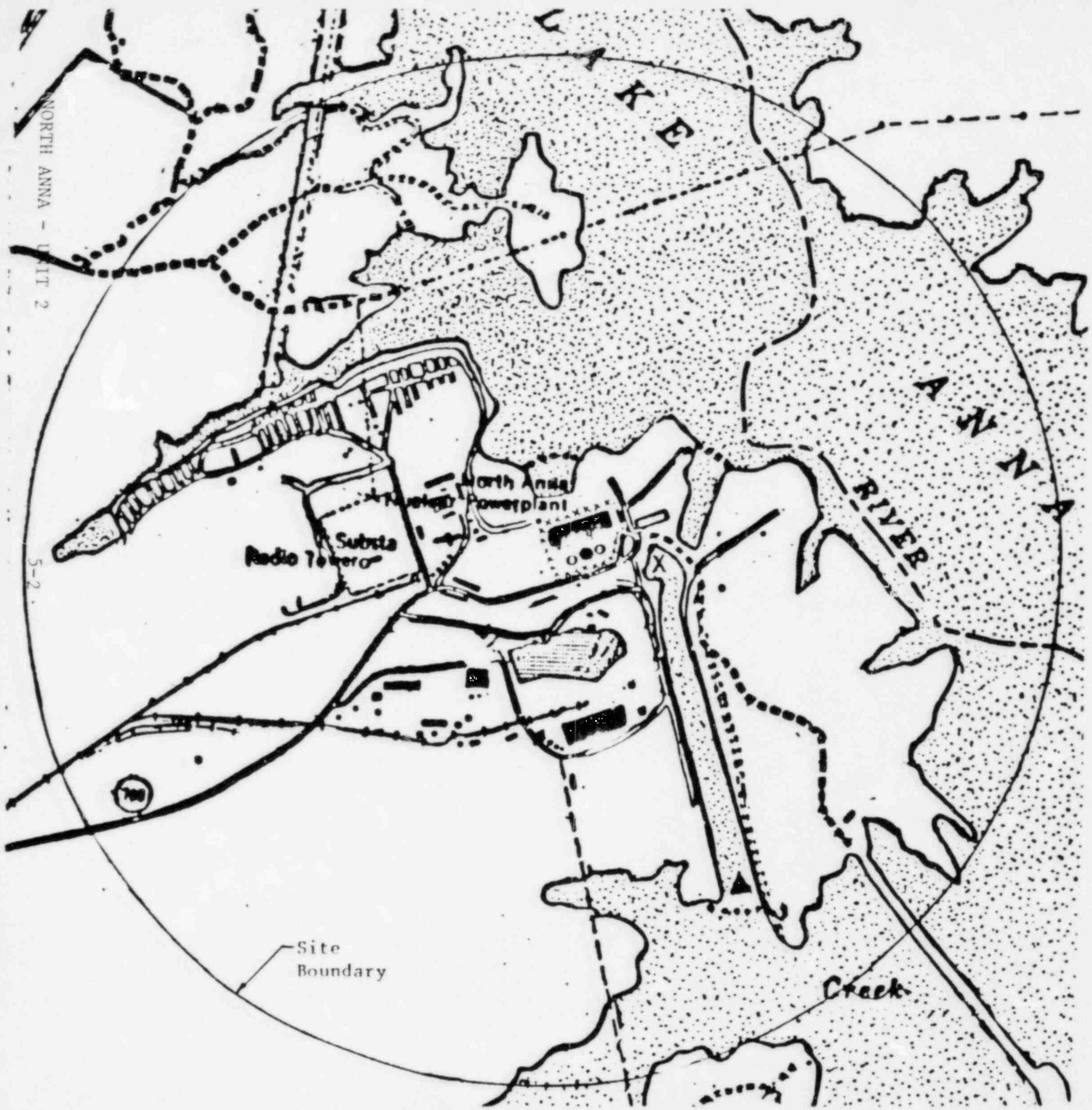
f. Nominal thickness of the cylindrial portion of the steel liner = 3/8 inches.

g. Net free volume = 1.825×10^6 cubic feet.

h. Nominal thickness of hemispherical dome portion of the steel liner = 1/2 inch.

DESIGN PRESSURE AND TEMPERATURE

5.2.2. The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.



● Gaseous Release
1. Process Vent- 157.5 ft.
2. Vent-Vent A&B and other release points considered ground level releases

X Liquid Release to the Discharge Canal

▲ Liquid Release to the Unrestricted Area

••• Buoy Barriers

xxxx Security Fence- Area outside is unrestricted for gaseous effluents

Land Maximum Member of the Public Occupancy = 336 hrs/year

Lake Maximum Member of the Public Occupancy = 2232 hrs/year

Figure 5.1-1

Map Defining Unrestricted Areas for Radioactive Gaseous and Liquid Effluents

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1 - 1971 for comparable positions and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, except for (1) the Supervisor - Health Physics 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.

6.4 TRAINING

6.4.1 The Station Manager is responsible for ensuring that retraining and replacement training programs for the facility staff are maintained and that such programs meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1 - 1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the SES.

6.5 REVIEW AND AUDIT

6.5.1 STATION NUCLEAR SAFETY AND OPERATING COMMITTEE (SNSOC)

FUNCTION

6.5.1.1 The SNSOC shall function to advise the Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The SNSOC shall be composed of the:

Chairman:	Station Manager
Vice Chairman:	Assistant Station Manager
Member:	Superintendent-Operations
Member:	Superintendent-Maintenance
Member:	Superintendent-Technical Services
Member:	Supervisor-Health Physics

ALTERNATES

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

MEETING FREQUENCY

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

QUORUM

6.5.1.5 A quorum of the SNSOC consists of the Chairman or Vice-Chairman and two members including alternates.

RESPONSIBILITIES

6.5.1.6 The SNSOC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8.1 and changes thereto, 2) all programs required by Specification 6.8.4 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Station Manager to affect nuclear safety.

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

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

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

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

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

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

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Station Nuclear Safety and Operating Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Station Nuclear Safety and Operating Committee.

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

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

AUTHORITY

6.5.1.7 The SNSOC shall:

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

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

c. Provide written notification within 24 hours to Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control of disagreement between the SNSOC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Manager-Nuclear Operations and Maintenance and the Director-Safety Evaluation and Control.

6.5.2 SAFETY EVALUATION AND CONTROL (SEC)

FUNCTION

6.5.2.1 SEC shall function to provide independent review of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Administrative controls and quality assurance practices

i. Other appropriate fields associated with the unique characteristics of the nuclear power plant

COMPOSITION

6.5.2.2 The SEC staff shall be composed of the Director-Safety Evaluation and Control and a minimum of three individuals who are qualified as staff specialists. Each SEC staff specialist shall have an academic degree in an engineering or physical science field and, in addition, shall have a minimum of five years technical experience in one or more areas given in Specification 6.5.2.1. These staff specialists shall not be directly involved in the licensing function.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the Director-Safety Evaluation and Control to provide expert advice to the SEC.

MEETING FREQUENCY

6.5.2.5 The SEC staff shall meet at least once per calendar month for the purpose of fostering interaction of reviews regarding safety-related operational activities.

REVIEW

6.5.2.7 The following subjects shall be reviewed by SEC:

a. Written safety evaluations of changes in the stations as described in the Safety Analysis Report, changes in procedures as described in the Safety Analysis Report and tests or experiments not described in the Safety Analysis Report which are completed without prior NRC approval under the provisions of 10 CFR 50.59(a)(1). This review is to verify that such changes, tests or experiments did not involve a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2) and is accomplished by review of minutes of the Station Nuclear Safety and Operating Committee and the design change program.

b. Proposed changes in procedures, proposed changes in the station, or proposed tests or experiments, any of which may involve a change in the Technical Specifications or an unreviewed safety question as defined in 10 CFR 50.59(a)(2). Matters of this kind shall be referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee following its review prior to implementation.

REVIEW (Cont'd)

c. Changes in the Technical Specifications or license amendments relating to nuclear safety prior to implementation except in those cases where the change is identical to a previously reviewed proposed change.

d. Violations and reportable occurrences such as:

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

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

3. Reportable occurrences as defined in the station Technical Specification 6.9.1.8.

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

e. The Quality Assurance Department audit program at least once per 12 months and audit reports.

f. Any other matter involving safe operation of the nuclear power stations which a duly appointed subcommittee or committee member deems appropriate for consideration, or which is referred to the Director-Safety Evaluation and Control by the Station Nuclear Safety and Operating Committee.

g. Reports and meeting minutes of the Station Nuclear Safety and Operating Committee.

AUTHORITY

6.5.2.9 The Director-Safety Evaluation and Control shall report to and advise the Manager-Nuclear Technical Services, who shall advise the Vice President-Nuclear Operations on those areas of responsibility specified in Section 6.5.2.7.

RECORDS

6.5.2.10 Records of SEC activities required by Section 6.5.2.7 shall be prepared and maintained in the SEC files and a summary shall be disseminated as indicated below each calendar month.

1. Vice President-Nuclear Operations

2. Nuclear Power Station Managers

3. Manager-Nuclear Operations and Maintenance

4. Manager-Nuclear Technical Services

5. Executive Manager-Quality Assurance *

6. Others that the Director-Safety Evaluation and Control may designate.

6.5.3 QUALITY ASSURANCE DEPARTMENT

FUNCTION

6.5.3.1 The Quality Assurance Department shall function to audit station activities. These audits shall encompass:

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

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

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

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

e. The Station Emergency Plan and implementing procedures at least once per 12 months.

f. The Station Security Plan and implementing procedures at least once per 12 months. *

g. Any other area of facility operation considered appropriate by the Executive Manager-Quality Assurance or the Senior Vice President-Power Operations.

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

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

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

* Pending NRC approval.

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

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

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

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

AUTHORITY

6.5.3.2 The Quality Assurance Department shall report to and advise the Executive Manager-Quality Assurance, who shall advise the Senior Vice President-Power Operations on those areas of responsibility specified in Section 6.5.3.1.

RECORDS

6.5.3.3 Records of the Quality Assurance Department audits shall be prepared and maintained in the department files. Audit reports shall be disseminated as indicated below:

1. Nuclear Power Station Manager
2. Manager-Nuclear Operations and Maintenance
3. Manager-Nuclear Technical Services
4. Director-Safety Evaluation and Control *
5. Supervisor of area audited
6. Nuclear Power Station Manager Quality Assurance

* Pending NRC approval.

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 SNSOC and submitted to the Director-Safety Evaluation and Control and the Manager-Nuclear Operations and Maintenance.

6.7 SAFETY LIMIT VIOLATION

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

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

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager-Nuclear Operations and Maintenance, and the Director-Safety Evaluation and Control shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the SNSOC. 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 Director-Safety Evaluation and Control and the Manager-Nuclear Operations and Maintenance 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, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program Implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

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.

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

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

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

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

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation spray, safety injection, chemical and volume control, gas stripper, and hydrogen recombiners. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements and

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

b. In-Plant Radiation Monitoring

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

(i) Training of personnel,

(ii) Procedures for monitoring, and

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

c. Secondary Water Chemistry

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

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

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

(iii) Identification of process sampling points,

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

(v) Procedures defining corrective actions for all control point chemistry conditions,

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

(vii) Monitoring of the condensate at the discharge of the condensate pumps for evidence of condenser inleakage. When condenser in-leakage is confirmed, the leak shall be repaired, plugged, or isolated within 96 hours.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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

STARTUP REPORTS

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (a) 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.

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

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

ANNUAL REPORTS[1/]

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

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

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

b. The complete results of the steam generator tube inservice inspections performed during the report period (Reference Specification 4.4.5.5.b.).

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety 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 Regional Office, of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

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

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

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

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

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

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

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

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to copy with accidents analyzed in the SAR.

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

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

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

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

THIRTY-DAY WRITTEN REPORT

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of 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. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

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

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

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

CORE SURVEILLANCE REPORT

6.9.1.10 The F_{xy} limit for Rated Thermal Power (F_{xy}^{RTP}) in all core planes containing Bank "D" control rods and in all unrodded core planes, the surveillance power level, P_m, for Technical Specifications 3.2.1 and 3.2.6, and the F_Q flyspeck basis shall be provided to the Director of the Regional Office of Inspection and Enforcement, with a copy to;

Director, Office of Nuclear Reactor Regulation
Attention: Chief of Core Performance Branch
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

at least 60 days prior to cycle initial criticality. In the event that the limits would be submitted at some other time during core life, they shall be submitted 60 days prior to the date the limits would become effective unless otherwise exempted by the Commission.

Any additional information needed to support the F_{xy}^{RTP} AND P_m submittal will be by request from the NRC and need not be included in this report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

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

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

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position,

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

Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 4.12-1 and discussion of all analyses in which the LLD required by Table 4.12-3 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**

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

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

The Radioactive Effluent Release Report shall be submitted within 60 days after January 1 of each year. This report shall include an assessment of the radiation doses to the maximum exposed MEMBERS OF THE PUBLIC due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. Annual meteorological data collected over the previous year shall be in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This meteorological data shall be retained in a file on site and shall be made available to the NRC upon request. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in the OFFSITE DOSE CALCULATION MANUAL (ODCM). Concurrent meteorological conditions or historical annual average atmospheric dispersion conditions shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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

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

If the dose to the maximum exposed MEMBER OF THE PUBLIC due to the radioactive liquid and gaseous effluents from the station during the previous calendar year exceeds twice the limits of Specification 3.11.1.2a, 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 the dose assessment shall include the contribution from direct radiation. The dose to the maximum exposed MEMBER OF THE PUBLIC shall show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation.

The Radioactive Effluent Release Reports shall include a list of unplanned releases as required to be reported in Technical Specification 6.9.1.9.e from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator, Region II, within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirement of the applicable reference specification:

a. Inservice Inspection Program Reviews shall be reported within 90 days of completion. Specification 4.0.5

b. ECCS Actuation shall be reported within 90 days of the occurrence. The report shall describe the circumstances of the actuation and the total accumulated cycles to date. Specification 3.5.2 and 3.5.3.

c. With the primary coolant specific activity >1.0 uCi/gram DOSE EQUIVALENT I-131 or $>100/\bar{E}$ uCi/gram, a specific activity analysis shall be included in the REPORTABLE OCCURRENCE report required pursuant to Specification 6.9.1.7. The information requested in Specification 3.4.8 shall also be included in that report.

d. With sealed source or fission detector leakage tests revealing the presence of ≥ 0.005 microcuries of removable contamination submit a special report on an annual basis outlining the corrective actions taken to prevent the spread of contamination. Specification 4.7.11.1.3.

e. With the MTC more positive than 0 $\Delta k/k/°F$ submit a special report within the next 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. Specification 3.1.1.4.

f. For any abnormal degradation of the structural integrity of the reactor vessel or the Reactor Coolant System pressure boundary detected during the performance of Specification 4.4.10.1, an initial report shall be submitted within 10 days after detection and a detailed report submitted within 90 days after the completion of Specification 4.4.10.1.

g. For any abnormal degradation of the containment structure detected during the performance of Specification 4.6.1.6.2, an initial report shall be submitted within 10 days after completion of Specification 4.6.1.6.2. A final report, which includes (1) a description of the condition of the liner plate and concrete, (2) inspection procedure, (3) the tolerance on cracking and (4) the corrective actions taken, shall be submitted within 90 days after the completion of Specification 4.6.1.6.2.

h. Inoperable Fire Detection Instrumentation, Specification 3.3.3.7.

i. Inoperable Fire Suppression Systems, Specifications 3.7.14.1, 3.7.14.2, 3.7.14.3, 3.7.14.4 and 3.7.14.5.

6.10 RECORD RETENTION

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

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

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

e. Records of changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

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

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

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

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

c. Records of facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components identified in Table 5.9-1.

g. Records of reactor tests and experiments.

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

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

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

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

l. Records of meetings of the SNSOC.

m. Records of meetings of the System Nuclear Safety and Operating Committee to issuance of Amendment No. 30.

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 would include procedures effective at specified times and QA records showing that these procedures were followed.

o. Records of secondary water sampling and water quality.

p. Records for Environmental Qualification which are covered under the provisions of Paragraph 6.13.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

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

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

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

c. An individual qualified in the 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 Shift Supervisor on duty and/or the Plant Health Physicist.

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*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 in high radiation areas.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Licensee initiated changes to the PCP:

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

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

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SNSOC.

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

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

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

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by the SNSOC.

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

6.15 MAJOR CHANGES TO RADIOACTIVE SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the radioactive solid waste systems:

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

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

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

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

   d. An evaluation of the change, in quantity of solid waste that differs from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change, which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

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

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

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

2. Shall become effective upon review and acceptance by SNSOC.

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

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2.0 LIMITING CONDITIONS FOR OPERATION

2.1 NON-RADIOLOGICAL - None

observations in a form consistent with National Weather Service procedures. Summaries of all data and observations shall be available to the NRC upon request.

Any modification to the onsite meteorological monitoring program as described above, or planned alterations of the area in the vicinity of the meteorological tower(s) that would interfere with the measurement of meteorological conditions representative of the site, will require written approval in accordance with Section 5.6.3.

Bases

The collection of meteorological data at the plant site will provide information which will be used to develop atmospheric diffusion parameters to estimate potential radiation doses to the public resulting from actual routine or abnormal releases of radioactive materials to the atmosphere, and to assess the actual impact of the plant cooling system on the atmospheric environment of the site area. A meteorological data collection program as described above is necessary to meet the requirements of subparagraph 50.36(a)(2) of 10 CFR Part 50, Appendix E to 10 CFR Part 50, and 10 CFR Part 51.

5.5.3 Procedures for Environmental Surveillance - Nonradiological

Not applicable.

5.5.4 Quality Assurance of Program Results

The procedures document shall provide for assurance of the quality of program results, including analytical measurements. This portion of the procedures document shall document the program in policy directives, designate a responsible organization or individuals, include purchased services (e.g., contractual lab or other contract services), include audits by licensee personnel, and include procedures for revising programs, systems to identify and correct deficiencies, investigate anomalous or suspect results, and review and evaluate program results and reports.

5.5.5 Changes in Procedures, Station Design or Operation

Changes in procedures, station design or operation may be made subject to conditions described below, provided such changes are approved by the SNSOC (Review and Audit responsibility per Section 5.3).

a. The licensee may (1) make changes in the station design and operation as described in the FES, FES Addendum and the Environmental Report, (2) make changes in the procedures described in the document developed in accordance with Subsection 5.5, and (3) conduct tests and experiments not described in the document developed in acordance with Subsection 5.5, without prior Commission approval, unless the proposed change, test or experiment involves a change in the objectives of the ETS, an unreviewed environmental question of substantive impact, or affects the requirements of Subsection 5.5.6 of these ETS.

b. A proposed change, test, or experiment shall be deemed to involve an unreviewed environmental question (1) if the probability of magnitude of environmental impact may be increased; or (2) if a possibility for a substantive environmental impact of a different type than any evaluated previously in the FES or FES Addendum may be created.

c. The licensee shall maintain records of changes in procedures and in facility design or operation made pursuant to this Subsection, to the extent that such changes constitute changes in procedures as described in the document developed in accordance with Section 5.5 or in the FES, FES Addendum and ER. The licensee shall also maintain records of tests and experiments carried out pursuant to paragraph "a" of this Subsection. These records shall include a written evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question of substantive impact or constitute a change in the objectives of these ETS, or affect the requirements of Section 5.5.6 of these ETS. The licensee shall furnish to the Commission, annually or at such shorter intervals as may be specified in the license, a report containing descriptions, analyses, interpretations, and evaluations of such changes, tests and experiments.

d. Changes in the procedures developed in accordance with Subsection 5.5 which affect sampling frequency, location, gear, or replication shall be reported to the NRC within 30 days after their implementation. These reports shall describe the changes made, the reasons for making the changes, an evaluation of the environmental impact of these changes, and the statement required under the provisions of Subsection 5.5.6.

5.5.6 Consistency with Initially Approved Programs

Any modifications or changes of the procedures developed in accordance with Subsection 5.5 must be governed by the need to maintain consistency with previously used procedures so that direct comparisons of data are technically valid. Such modifications or changes must be justified and supported by adequate comparative sampling programs or studies demonstrating the comparability of results or which provide a basis for making adjustments that would permit direct comparisons.

These demonstrations of comparability shall be submitted to the NRC in accordance with the provision of Subsection 5.5.5 and 5.6.1 of these ETS.

5.6 Station Reporting Requirements

5.6.1 Routine Reports - None.

5.6.2 Nonroutine Reports

5.6.2.1 Nonroutine Non-Radiological Environmental Operating Report

None.

5.6.3 Changes in Environmental Technical Specifications

A report shall be made to the NRC prior to implementation of a change in plant design, in plant operation, or in procedures described in Section 5.5 if the change would have a significant effect on the environment or involves an environmental matter or question not previously reviewed and evaluated by the NRC. The report shall include a description and evaluation of the change and a supporting benefit-cost analysis.

Request for changes in Environmental Technical Specifications shall be submitted to the Director, Office of Nuclear Reactor Regulation, for review and authorization. The request shall include an evaluation of the environmental impact of the proposed changes and a supporting benefit-cost analysis.

5.6.4 Changes in Permits and Certifications

None

5.7 Records Retention

Records and logs relative to the following areas shall be made and retained for the life of the station:

a. Records and drawings detailing plant design changes and modifications made to systems and equipment as described in Section 5.6.3.

b. Reports from environmental monitoring, surveillance, and special surveillance and study activities required by these Environmental Technical Specifications.

ATTACHMENT 2

JUSTIFICATIONS FOR DEVIATING FROM

NUREG-0472 REVISION 3

In Table 3.3-12, the Steam Generator Blowdown Effluent Line and the Turbine Building Sumps Effluent Line are not included because they are not release points. The justification for sampling the turbine building sump pumps whenever the secondary coolant activity exceeds $1x10^{-5}$ uCi/ml has been evaluated. It is well documented that in a recirculating U-tube steam generator, the nonvolatile radionuclides and most of the iodine leaking from the primary coolant concentrate in the liquid phase in the steam generator. From North Anna Power Station Units 1 and 2 Updated Final Safety Analysis Report, Virginia Electric and Power Company, Section 15.4.3.4.2, assuming equilibrium is reached between the liquid and vapor iodine concentrations, the effective partition factor is $1x10^{-2}$. This partition factor is recommended in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)", and WCAP-8253," Source Term Data for Westinghouse Pressurized Water Reactors". In addition, NUREG-0017 recommends a partition factor of $1x10^{-3}$ for Cs, Rb, and other nuclides.

Applying the $1x10^{-2}$ iodine partition factor to a secondary coolant concentration of $1x10^{-5}$ uCi/ml, an I-131 concentration of $1x10^{-7}$ uCi/ml would be expected in the turbine building sump liquid effluent. For Cs, Rb and other nuclides, the concentration would be $1x10^{-8}$ uCi/ml. From Table 4.11-1 in NUREG-0472, Rev. 3, the lower limit of detection (LLD) for I-131 in liquid effluents is $1x10^{-6}$ uCi/ml and $5x10^{-7}$ uCi/ml for principal gammer emitters. The turbine building sump concentration, based on a secondary coolant concentration of $1x10^{-5}$ uCi/ml, would be less than the NUREG-0472, Rev. 3, LLD's. As an indicator for turbine building sump concentration, secondary coolant concentration of $1x10^{-5}$ uCi/ml would be conservative. Basing turbine building sump sampling on secondary coolant concentration would eliminate the unnecessary burden of sampling and analyzing samples with concentrations below LLD's representing insignificant fractions of total station liquid effluent release activity.

The only release point for continuous composite samplers and sampler flow rate monitors is on the Clarifier Effluent Line. The Discharge Tunnel is not included because it is not a release point. There are currently no radioactivity recorders on the Liquid Radwaste Effluent Line because it does not provide alarm/setpoint function for release termination or on the Steam Generator Blowdown Effluent Line because it is not a release point at North Anna.

Action 28, in Table 3.3-12 of NUREG-0472, Revision 3, has been deleted because it pertains to batch releases and the Liquid Radwaste Effluent Line has a continuous release. Action 26 therefore replaces Action 28. Action 29, of NUREG-0472, Revision 3, has been deleted because the Steam Generator Blowdown Effluent Line has been deleted as a separate release point. All action statements have been renumbered so they are numbered sequentially in the Technical Specifications.

Table 4.3-12 has been revised to be consistant with Table 3.3-12. Table notation 3, of NUREG-0472, Revision 3, is deleted because North Anna 1 and 2 are operating units and have established calibration procedures. The circuit failure and downscale failure have been deleted from table notation 1 and 2, of NUREG-0472, Revision 3, because it would cost approximately $315,000 to backfit monitors on the Liquid Radwaste Effluent Line, Service Water System Effluent Line and Circulating Water System Effluent Line. If Vepco were to backfit monitors on these effluent lines, it would take approximately one year from the start of engineering. Vepco believes it is not benificial to backfit on these effluent lines. In addition, a channel check is performed every shift by the operator to determine if there is a downscale failure and circuit failure on the monitors. If the monitor reads zero on the scale, the operator refers to his procedure to take action.

In Table 3.3-13, the Waste Gas Holdup System Explosive Gas Monitoring System currently has only one hydrogen monitor and one oxygen monitor. To install another hydrogen and oxygen monitor, as required by NUREG-0472, Revision 3, there would be a one year lead time from the start of engineering work to final installation and the backfit would also cost approximately $75,000. Vepco believes it is not beneficial to do this backfit. The iodine sampler, particulate sampler and sampler flow rate monitor have been deleted (from NUREG-0472, Revision 3) from the Condenser Air Ejector System because moisture will wipe-out the charcoal system. It will not operate on a continuous basis but grab samples are obtained on a daily basis.

All samplers and monitors listed in Table 3.3-13 are shared between Units 1 and 2. The Containment Purge System has been deleted, from NUREG-0472, Revision 3 because it uses the Ventilation Vent System. The Containment Purge System is not a separate release point. The Auxiliary Building Ventilation System, Fuel Storage Area Ventilation System and Radwaste Area Ventilation System (of NUREG-0472, Revision 3) have been deleted because they are part of the Ventilation Vent System. The Steam Generator Blowdown Vent System has been deleted because it has vent condensers. The action statements have been revised to reflect the monitoring and sampling frequencies.

Table 4.3-13 has been made consistent with Table 3.3-13. Table notations 1 and 2 (of NUREG-0472, Revision 3) have been changed for the reasons discussed in Table 4.3-12. Table notation 3 (of NUREG-0472, Revision 3) has been deleted because Units 1 and 2 are operating and have calibration procedures. A footnote has been added to the sampler flow rate monitor on the Process Vent System to ensure channel check consists of verifying indications of flow during periods of releases and that a channel check shall be made at least once per 24 hours on continuous, periodic or batch releases. The circuit failure and downscale failure have been deleted for reasons discussed above.

In Table 4.11-1, a footnote has been added under batch releases for whenever the secondary coolant activity exceeds 10^{-5} uCi/ml the turbine building sump pumps shall be placed in manual operation and samples shall be taken and analyzed prior to release. Secondary coolant activity samples shall be collected and analyzed on a weekly basis. These samples are analyzed for gross activity or gamma isotopic activity within 24 hours. The Clarifier System is open to the atmosphere and the effluent is monitored continuously by a radiation monitor and composite sampler.

Throughout RETS, all reference to any organ has been changed to the critical organ. Where this change is applied, a footnote has been added to state that the critical organ is addressed in the ODCM. The reference to the critical organ has been performed to provide clarity.

In Specification 3.11.1.2, of NUREG-0472, Revision 3, the reference to drinking water has been deleted because the nearest drinking water supply is in Doswell, Virginia which is located approximately 20 miles from North Anna Power Station.

In Specification 3.11.1.3 of NUREG-0472, Revision 3, the liquid radwaste treatment system has been changed to the liquid radwaste ion exchanger system because the liquid radwaste treatment system was a very broad area and certain components could be interpreted to be in that system and they should not.

In Table 4.11.2, the Process Vent System and Vent. Vent A and B were listed as release points where gaseous effluents are discharged from the facility (Section C). In this sampling analysis frequency, footnotes c and d (from NUREG-0472, Revision 3) were deleted. This is because the amounts of tritium that might be detected are very small since North Anna Units 1 and 2 have been operating for several years. Monthly ventilation vent B tritium grab sample data were reviewed for the years 1980, 1981, and 1982 to date, along with containment purge tritium grab sample data for the years 1980, 1981, and 1982 to date. The review of sample data has not indicated any significant increases in tritium activity in ventilation vent B monthly tritium grab samples during periods of containment purge. Each ventilation vent B monthly tritium grab sample remained below the technical specification LLD of $1x10^{-6}$ uCi/ml. In addition, only 3% of the containment purge tritium grab samples were above the technical specification LLD for tritium. Therefore, it has been determined that the additional sampling requirements of footnotes c and d in Table 4.11-2, NUREG-0472, Rev. 3 are unwarranted. A separate category has been created for the Condenser Air Ejector Vent and Steam Generator Blowdown Vent. In reference to the Condenser Air Ejector Vent we say we will obtain a weekly grab sample. This weekly grab sample will be taken only when there is primary to secondary leakage. On the Steam Generator Blowdown Vent we say that a weekly grab sample will be taken. This grab sample will be looking for radiogases. The Steam Generator Blowdown Tank Vent Condenser will condense the iodine in the steam and it will then return to the steam generator blowdown stream for discharge through the clarifiers. The reference to I-131 and other isotopes has been deleted from the continuous, particulate sample. Note b, on page 3/4 11-12 which relates to the type of activity analysis does not exclude I-131 and other isotopes. I-131 and other isotopes will be reported if the LLD reaches 1×10^{-11} uCi/ml.

In Specification 4.11.2.6 of NUREG-0472, Revision 3 the determination of the quantity of radioactive material in each gas storage tank has been changed from at least once per 24 hours to once per month when radioactive materials are being added to the tank. The noble gas activities which would result in the Waste Gas Decay Tank for the limiting conditions specified in Attachment 3 indicate that a monthly sampling frequency would ensure the maximum activity of the Waste Gas Decay Tank is within the 24,000 Curie limit.

In Specification 3.11.3, the LCO has been changed to clarify how solidification pertains to the process control program.

Tables 3.12-1 and 3.12-2 have been renumbered Tables 4.12-1 and 4.12-2. These tables seem to be surveillance requirements rather than limiting conditions for operation. Table 4.12-1, from revision 3, has been renumbered 4.12-3. North Anna Power Station has 36 routine monitoring locations that sample direct radiation.

For North Anna Power Station, the highest D/Q is SE, NNE, ESE (Site Boundary) for airborne radiation. Sectors SE, ESE, & NNE are unavailable for airborne sample locations due to power source restrictions, and/or site boundary located over water. Sector SSE, which is adjacent to SE sector, has an airborne sampling station. Sectors ESE and NNE have an airborne sampling station located within 6-km of the station, which is very accessible to sample. Currently, North Anna has 3 site boundary sample locations at sectors SSE, WSW, and WNW.

For North Anna Power Station, the exposure pathway or sample for waterborne radiation is obtained through monthly grab surface water samples from the circulating water discharge. These samples are currently obtained for monthly gamma isotopic analysis. Composites of each sample are analyzed quarterly for tritium. Historically, no increase in activity has been detected from surface water samples with exception of tritium, which is expected.

Drinking water is not presently sampled. The preoperational environmental sampling program included drinking water. At the conclusion of preoperational environmental sampling program, it was determined that drinking water samples could be terminated.

The exposure pathway or sample of milk for North Anna Power Station with the highest D/Q is in the SSW, N, and E. There are no dairies located within 5-km in sectors SSW, N, and E. In sectors SSW and S, within 8-km of the station, dairy milk samples are obtained. One control milk sample is obtained in sector NW approximately 11-km from the site, which is a least prevalent wind direction. All milk samples are obtained on a monthly basis. Historically, based on sample analysis, monthly sampling has been deemed appropriate.

Presently, bass, catfish and sunfish are sampled on an annual basis at both upstream and downstream locations. These are the recreationally most important species in the vicinity of plant discharge area.

The exposure pathway or sample of food products has been deleted from NUREG-0472, Revision 3. No irrigation is utilized in the area of the lake. Presently, sampling of an edible broad leaf vegetation (cabbage), which is located in sectors S and ENE is being performed at harvest time.

Table notation f, of NUREG-0472, Revision 3, has been deleted because the surface waterborne samples will be taken at the circulating water discharge. Table notation h, of NUREG-0472, Revision 3, has been deleted because the ground water samples are taken for drinking and irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination. North Anna Power Station currently has no irrigation in the area. Drinking water samples were terminated due to the Preoperational Environmental Sampling Program. Table notation d in Table 4.12-3, of NUREG-0472, Revision 3, has been deleted and the LLD for I-131 in water has been added.

In Specification 3/4 12.2, a dose commitment of 25 percent or greater is used instead of 20 percent or greater so that North Anna can have additional margin. It is also understood that the 20 percent is an arbiturary number. North Anna Power Station does not have elevated releases, therefore, all reference to elevated releases has been deleted from NUREG-0472, Revision 3.

In Specification 6.9.1.12 (of NUREG-0472, Revision 3) the reference to 10 CFR Part 61 information has been deleted. The reason for this is that 10 CFR Part 61 is a proposed rule and this information is already reported in the requirements of Regulatory Guide 1.21.

The Specification for reporting major changes to radioactive liquid gaseous and solid waste treatment systems has been changed to address radioactive solid wastes only. The portion related to liquid and gaseous wastes currently appears in North Anna Unit 1 License Condition 2.D(3)(e) and North Anna Unit 2 License Condition 2.G.

All reference to radiological effluents, radiological environmental monitoring and radiological reporting requirements have been deleted from the Appendix B Technical Specifications. The reason is due to the fact that this information is now being incorporated into the Appendix A Technical Specifications.

ATTACHMENT 3

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION

HEALTH PHYSICS

JUSTIFICATION OF MONTHLY

SAMPLING FOR

WASTE GAS DECAY TANKS

DECEMBER 1, 1982

A. Subject: Justification for monthly sampling of Waste Gas Decay Tanks at North Anna Power Station.

B. References:

1. North Anna Power Station Environmental Technical Specifications, Copy No. 38, Section 2.2.3.
2. North Anna Power Station Updated Final Safety Analysis Report, Chapter 11.
3. North Anna Power Station Emergency Plan Implementing Procedure, EPIP 4.09, Source Term Assessment, Revision 2.
4. North Anna Power Station Technical Specifications-Unit 1, Copy 41.
5. USAEC Technical Information Document, TID 14844, March 23, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites".

C. Introduction:

Reference (1) states: "The maximum activity to be contained in one Waste Gas Decay Tank (WGDT) shall not exceed 25,000 Curies (considered as Xe-133)." A justification is written for sampling the WGDT monthly. The physical limitations of the tank, i.e. design pressure and design volume coupled with the activity of the noble gases available to the WGDT, will ensure that the quantity of radioactive material in each WGDT will be $\leq$ 25,000 curies.

D. Approach:

Reference (2) was used to perform a design analysis of the Waste Gas Decay Tanks and to define the technical parameters associated with the accumulation of radiogas in these tanks. Using assumptions provided in reference (2), a quantitative assessment of noble gas and halogen inventory was established under conditions of 1.0 percent failed fuel in the primary reactor coolant. An alternate review of FSAR data and assumptions, using the maximum activities for radiogas under conditions of 0.2 percent failed fuel and expected operating parameters indicate that the Technical Specifications contained in reference (2) would not be exceeded. Employing the equations and dose conversion factors given in reference (3), the dose equivalent Xe-133 was determined; the Environmental Tech. Specs. define the total activity accumulated in the WGDT to be considered Xe-133. The DOSE EQUIVALENT I-131 was then calculated by multiplying the halogen activities in the primary reactor coolant by dose conversion factors listed in Table III of reference (5). The DOSE EQUIVALENT I-131 was quantitatively compared with the Tech. Spec. given in reference (4) for allowable specific activity of primary coolant. A percent deviation between these two concentration levels was calculated. A second value for DOSE EQUIVALENT Xe-133 was determined using the Xe/I ratio reflecting the percent difference between the DOSE EQUIVALENT I-131 quantities.

The total amount of radiogas in the WGDT under design criteria and expected maximum activities was calculated using a tank volume of 462 cu. ft. and a maximum tank pressure of 175 psig.

E. Data and Calculations:

Quantitative assessment of the Waste Gas Decay Tanks was based on a set of conservative assumptions provided in reference (2) including the following:

1. Parameters listed in reference (2) Table 11.1-5.

   "A comparison of the gap activities for a 15 X 15 and 17 X 17 fuel assembly shows that the core gap activities for the 17 X 17 fuel assembly is approximately one-half of those for the 15 X 15 fuel assembly. This is the result of the lower operating temperatures and therefore slower diffusion rate in the 17 X 17 fuel."

2. 1.0 percent Failed Fuel.
3. 100 percent noble gases available to the WGDT.
4. 25 percent halogens available to WGDT.
5. Defective fuel rods were present at the initial core loading and were uniformily distributed throughout the core.
6. Average temperature of reactor coolant system.
7. Coolant density correction of 1.35 was made to obtain the correct activities downstream of regenerative heat exchanger.

Assumption for Maximum Radioactive Activites in Waste Gas Decay Tank-Expected Case:

1. Calculation performed for 2 units operating in base load cycle.
2. 0.2 percent Failed Fuel.
3. One unit operating 4 weeks behind the second unit.
4. 300 day feed including the stripping of all noble gases from one unit at the end of the fuel cycle.
5. 60 day decay after feed cycle is complete.
6. 10 day bleed cycle for WGDT.
7. Stripping of all noble gases from the second unit and the feeding of this gas to the tank at the beginning of the feed cycle.
8. All of the noble gases and 0.1 percent iodines in the letdown will be removed at the gas stripper and sent to the WGDT.

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT
FISSION PRODUCT ACTIVITIES
(15 X 15 FUEL ASSEMBLY)

Parameter	Value
Core thermal power, max. calculated, MWT	2,900
Fraction of fuel containing clad defects	
Design	0.01
Expected	0.002
Reactor coolant liquid volume, including pressurizer, ft^3	9,380
Reactor coolant average temperature, ^oF	577
Purification flow rate (normal), gpm	60
Effective cation demineralizer flow, gpm	6.0
Volume control tank volumes, ft^3	
Vapor	180
Liquid	120
Fission product escape rate coefficients, sec^{-1}	
Noble gas isotopes	6.5×10^{-8}
Br, I, and Cs isotopes	1.3×10^{-8}
Te isotopes	1.0×10^{-9}
Mo isotopes	2.0×10^{-9}
Sr and Ba isotopes	1.0×10^{-11}
Y, La, Ce, and Pr isotopes	1.6×10^{-12}
Mixed bed demineralizer decontamination factors	
Noble gases and Cs-134, Cs-136, Cs-137, Y-90, Y-91, and Mo-99	1.0
All other isotopes	10.0
Cation bed demineralizer decontamination factor for Cs-134, Cs-136, Cs-127, Y-90, Y-91, and Mo-99	10.0

ISOTOPE	*CONCENTRATION (uCi/cc)	DOSE CONVERSION FACTOR	Xe-133 EQUIVALENT
I-131	4.6E-1	8.57	3.94
I-132	1.7E-1	5.13E+1	8.72
I-133	7.5E-1	1.37E+1	1.03E+1
I-134	1.04E-1	5.91E+1	6.15
I-135	4.03E-1	3.55E+1	1.43E+1
Kr-85	3.86	5.03E-2	1.94E-1
Kr-85m	1.59	3.55	5.64
Kr-87	9.2E-1	1.78E+1	1.64E+1
Kr-88	2.78	4.56E+1	1.27E+2
Xe-133	2.13E+2	1.0	2.13E+2
Xe-133m	2.36	9.09E-1	2.15
Xe-135	4.62	5.57	25.7
Xe-135m	1.43E-1	9.74	1.39
Xe-138	5.08E-1	2.54E+1	1.29E+1

* Quantities available to the Waste Gas Decay Tanks.

Σ Xe-133 = 4.48E+2 uCi/cc. Dose Equivalent Xe-133

I-131	4.6E-1	1.00	4.6E-1
I-132	1.7E-1	3.61E-2	6.14E-3
I-133	7.5E-1	2.70E-1	2.03E-1
I-134	1.04E-1	1.69E-2	2.37E-3
I-135	4.03E-1	8.38E-2	3.38E-2

Σ I-131 = 7.05E-1 uCi/cc. Dose Equivalent I-131

Xe/I = 4.48E+2/7.05E-1 uCi/cc = 6.35E+2

In accordance with reference (4), the specific activity of the primary coolant shall be limited to: $\leq$ 1.0 uCi/gm DOSE EQUIVALENT I-131.

The percent deviation between the calculated DOSE EQUIVALENT I-131 and Tech. Spec. for specific activity in primary reactor coolant is 29.5%.

Xe/I ratio = .295 X 6.35E+2
= 1.87E+2

1.87E+2 X 7.05E-1 uCi/cc. = 1.32E+2 uCi/cc. DOSE EQUIVALENT Xe-133

1.32E+2 uCi/cc. X 462 ft^3 X 2.832E+4 cc/ft^3 X 175 psig X 6.804E-2 atm. X 1.0E-6 Ci/uCi
= 20,552 Ci

Radiogas (considered as Xe-133) accumulated in WGDT under design volume and maximum pressure is 20,552 Curies.

WASTE GAS DECAY TANK - EXPECTED CASE

MAXIMUM RADIOACTIVE GASEOUS ACTIVITY

ISOTOPE	CONCENTRATION (uCi/cc)	DOSE CONVERSION FACTOR	EQUIVALENT Xe-133
Kr-85m	8.9E-1	3.55	3.16
Kr-85	1.4E-2	5.03E-2	7.04E-4
Kr-87	7.2E-3	1.78E+1	1.28E-1
Kr-88	5.0E-1	4.56E+1	2.28E+1
Xe-131m	3.6E-1	4.3E-1	1.54
Xe-133m	7.4	9.09E-1	6.73
Xe-133	7.8E-2	1.0	7.8E-2
Xe-135m	3.1E-5	9.74	3.02E-4
Xe-135	5.4	5.57	30.08
Xe-136	4.8E-11	2.54E+1	1.22E-9
I-131	1.0E-3	8.57	8.57E-3
I-132	3.2E-5	5.13E+1	1.64E-3
I-133	9.3E-4	1.37E+1	1.27E-2
I-134	7.0E-8	5.91E+1	4.14E-6
I-135	2.4E-4	3.55E+1	8.52E-3

Σ Xe-133 = 6.45E+1 uCi/cc DOSE EQUIVALENT Xe-133

6.45E+1 uCi/cc x 462 ft^3 x 2.832E+4 cc/ft^3 x 175 psig x 6.804E-2 atm. x 1.0E-6 Ci/uCi = 10,048 Ci

Radiogas (considered as Xe-133) accumulated in WGDT under design volume and maximum pressure is 10,048 Curies.

F. <u>Conclusion:</u>

The correlation between DOSE EQUIVALENT I-131 and DOSE EQUIVALENT Xe-133, both quantities calculated on the basis of operating parameters and design criteria, coupled with the expected values of WGDT noble gas inventory, demonstrate the adequacy of a monthly sampling frequency for each waste gas storage tank to ensure the quantity of radioactive material contained in each tank is $\leq$ 25,000 Curies considered as Xe-133.

Under conditions which produce specific activities of primary reactor coolant exceeding 1.0 uCi/gm, it will be necessary to sample the WGDT once per 24 hours.

Based upon the justification presented in this report, it is recommended that the surveillance requirements for waste gas Storage Tanks contained in the Radiological Effluent Technical Specification (Section 4.11.2.6) be revised as follows:

"The quantity of radioactive material contained in each Waste Gas Decay Tank shall be determined to be within the above limit at least once a month when the specific activity of the primary reactor coolant is $\leq$ 1.0 uCi/gm DOSE EQUIVALENT I-131. Under conditions which result in a specific activity > 1.0 uCi/gm DOSE EQUIVALENT I-131, the Waste Gas Storage Tanks shall be sampled once per 24 hours."