number of operable channels and the number of channels when tripped will cause reactor trip.

1.7 INSTRUMENTATION SURVEILLANCE

1) CHANNEL CHECK

CHANNEL CHECK is a qualitative determination of acceptable operability by observation of channel behavior during operation. This determination shall include, where possible, comparison of the channel with other independent channels measuring the same variable or radioactive source check of the Area and Process Radiation Monitoring Systems for channels.

2) CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST consists of injecting a simulated signal into the channel to verify that it is operable, including alarm and/or trip initiating action.

3) CHANNEL CALIBRATION

CHANNEL CALIBRATION consists of the adjustment of channel output such that it responds, with acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including alarm or trip, and shall be deemed to include the CHANNEL FUNCTIONAL TEST.

1.8 SHUTDOWN

1) Cold Shutdown

The reactor is in the cold shutdown condition when the reactor is subcritical by at least 1% $\Delta k/k$ and T_{avg} is less than 200F.

2) Hot Shutdown

The reactor is in the hot shutdown condition when it

1.24 GAS DECAY TANK SYSTEM

The GAS DECAY TANK SYSTEM is 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.

1.25 VENTILATION EXHAUST TREATMENT SYSTEM

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

1.26 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM shall contain the provisions, based on full scale testing, to assure that dewatering of spent bead resins results in a waste form with the properties that meet the requirements of 10CFR61 (as implemented by 10CFR20) and of the low level radioactive waste disposal site at the time of disposal.

1.27 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints.

1.28 DOSE EQUIVALENT I-131

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", or in NRC Regulatory Guide 1.109, Rev. 1 October, 1977.

1.29 PURGE - PURGING

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

1.30 VENTING

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

1.31 SITE BOUNDARY

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

1.32 UNRESTRICTED AREA

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

1.33 MEMBER(S) OF THE PUBLIC

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

3.9 RADIOACTIVE MATERIALS RELEASE

Applicability: Applies to the controlled release of all liquid and gaseous waste discharge from the plant which may contain radioactive materials, and to the disposal of radioactive bead resins.

Objective: To establish conditions for the release of liquid and gaseous radioactive materials and radioactive bead resins, and to assure that all such releases are within the limits specified in 10CFR20. In addition, every reasonable effort shall be made to control the rate of release of radioactive materials in liquid and gaseous waste discharged from the site to UNRESTRICTED AREAS (See Figure 5.1-1) as low as is reasonably achievable.

Specifications:

1. LIQUID EFFLUENTS

a. Concentration

The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS shall not exceed the concentrations specified in 10CFR20, APPENDIX B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall not exceed 2×10^{-4} µCi/ml.

1. Radioactive liquid waste shall be sampled and analyzed according to the sampling and analysis program of Table 3.9-1.

2. The results of the radioactivity analysis 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 above limits.

3. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, restore the concentration to within the above limits.

b. Dose

The dose or dose commitment per reactor to a MEMBER OF THE PUBLIC from any radioactive materials in liquid effluent released to UNRESTRICTED AREAS shall be limited, during any calendar quarter, to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ.

1. Calculations shall be performed in accordance with methods described in the ODCM at least once per month to determine the cumulative dose commitment due to liquid effluents.

2. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit a report to the Commission within 30 days, pursuant to Specification 6.9.3.e.

c. Monitoring Instrumentation

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.9-2 shall be OPERABLE with their alarm/trip setpoints (if applicable) set to ensure that the limits of Specification 3.9.1.a are not exceeded. The setpoints shall be determined in accordance with methods described in the ODCM.

1. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by Specification 3.9.1.c, 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.

2. With less than the minimum number of radioactive liquid effluent monitors being OPERABLE, take the ACTION shown in Table 3.9-2.

3. Each radioactive liquid effluent monitor shall be demonstrated OPERABLE by performance of operations at the frequencies shown in Table 4.1-3.

d. Radwaste Treatment

Appropriate subsystems of the liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses, to UNRESTRICTED AREAS due to liquid effluents, when averaged monthly, would exceed 0.12 mrem to the total body or 0.4 mrem to any organ.

1. Doses due to liquid releases shall be projected at least once per month, in accordance with the ODCM, unless the liquid radwaste treatment system is being used.

2. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit a report to the Commission within 30 days pursuant to Specification 6.9.3.f.

TABLE 3.9-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING[g] FREQUENCY	MINIMUM[g] ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/ml)
A. Batch Waste Release Tanks[c]	PR Each Batch	PR Each Batch	Principal Gamma Emitters[b,h]	5×10^{-7}
			I-131[h]	1×10^{-6}
	PR One Batch/M	M	Dissolved and Entrained Gases[b]	1×10^{-5}
	PR Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	PR Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Plant Continuous Releases[e]				
1. Steam Generator Blowdown	$4/M^i$	$4/M^i$	Principal Gamma Emitters[b,h]	5×10^{-7}
			I-131[h]	1×10^{-6}
	M^f	M^f	Dissolved and Entrained Gases[b]	1×10^{-5}
	$4/M^f$	M^f Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	$4/M^f$	Q^f Composite[d]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Storm Drain	M	M	Principal Gamma Emitters[b,h]	5×10^{-7}
			I-131[h]	1×10^{-6}

TABLE 3.9-1 (Cont.)

TABLE NOTATION

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 3.9-1 (Cont.)

TABLE NOTATION

b. The principal gamma emitters for which the LLD limit specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99m, Cs-134, Cs-137, Ce-141, Ce-144, Xe-133, Xe-133m and Xe-135. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level. When a radionuclide's calculated LLD is greater than its listed LLD limit, the calculated LLD should be assigned as the activity of the radionuclide; or, the activity of the radionuclide should be calculated using measured ratios with those radionuclides which are routinely identified and measured.

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

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 which is representative of the liquids released. Prior to analysis, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

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

f. Sampling and analysis of steam generator blowdown for these species, is only necessary when primary to secondary leakage is occurring as indicated by the condenser air ejector monitor.

g. Frequency designations are defined in Table 4.1-1, Sheet 4.

h. If performance of an isotopic analysis is not possible, liquid releases may be made for up to 7 days on the basis of a gross beta-gamma analysis at an LLD level of 1×10^{-7} µCi/ml.

i. Sampling and analysis of steam generator blowdown is not required during COLD or REFUELING SHUTDOWNS.

TABLE 3.9-2.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release			
a. Liquid Radwaste Effluent Line	(1)	*	1
b. Steam Generator Blowdown Effluent Line	(1)	**	2
2. Flow Rate Measurement Devices			
a. Liquid Radwaste Effluent Line	(1)	*	3
b. Steam Generator Blowdown Effluent Line	(1)	**	3

* During liquid effluent releases.
** During blowdown operations.

Table 3.9-2 (Cont.)

TABLE NOTATION

ACTION 1 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Specification 3.9.1.a.1 and;

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valving (one performs, one verifies);

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 2 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 analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} µCi/ml or analyzed isotopically (gamma) at a limit of detection of at least 5×10^{-7} µCi/ml:

1. At least once per 12 hours when the specific activity of the secondary coolant is > 0.01 µCi/ml DOSE EQUIVALENT I-131.

2. At least once per 24 hours when the specific activity of the secondary coolant is $\leq$ 0.01 µCi/ml DOSE EQUIVALENT I-131.

ACTION 3 With the number of channels OPERABLE less than required by the Minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

2. GASEOUS EFFLUENTS

a. Dose Rate

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:

Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin due to noble gases and less than or equal to 1500 mrem/yr to any organ due to I-131, I-133, tritium and for all radioactive materials in particulate form with half lives greater than 8 days.

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

2. The release rate of radioactive materials, other than noble gases, in gaseous effluents shall be determined to be within the above limits in accordance with the methods described in the ODCM and by obtaining representative samples and performing analyses in accordance with the sampling and analysis program, specified in Table 3.9-3.

3. In the event the dose rate(s) exceeds the above limits, decrease the release rate to comply with the limit(s) given in Specification 3.9.2.a.

b. Dose - Noble Gases

The air dose per reactor to areas at and beyond the SITE BOUNDARY due to noble gases released in gaseous effluents shall be limited, during any calendar quarter, to $\leq$5 mrad for gamma radiation and $\leq$10 mrad for beta radiation.

1. Cumulative dose contributions for the current calendar quarter shall be determined in accordance with methods described in the ODCM at least once per month.

2. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit a report to the Commission within 30 days, pursuant to Specification 6.9.3.e.

c. Dose - I-131, I-133, Tritium and Particulates

The dose per reactor to a MEMBER OF THE PUBLIC, due to I-131, I-133, tritium and to particulates with half-lives greater than 8 days in airborne effluents released to areas at and beyond the SITE BOUNDARY shall not exceed 7.5 mrem to any organ during any calendar quarter.

1. Cumulative dose contributions for the current calendar quarter shall be determined in accordance with the ODCM at least once per month.

2. With the calculated dose from the release of I-131, I-133, tritium and materials in particulate form with half lives greater than 8 days, in gaseous effluents exceeding the above limit, prepare and submit a report to the Commission within 30 days, pursuant to Specification 6.9.3.e.

d. Monitoring Instrumentation

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.9-4 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.2.a are not exceeded. The setpoints shall be determined in accordance with methods described in the ODCM.

1. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by Specification 3.9.2.d, 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.

2. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels being OPERABLE, take the ACTION shown in Table 3.9-4.

3. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the channel operations at the frequencies shown in Table 4.1-4.

e. Radwaste Treatment

The GAS DECAY TANK SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge if the projected gaseous effluent dose per

reactor due to gaseous effluent releases to areas at and beyond the SITE BOUNDARY when averaged over a month exceeds 0.4 mrad for gamma radiation and 0.8 mrad for beta radiation, and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge if the projected gaseous effluent dose per reactor due to gaseous effluent releases to areas at and beyond the SITE BOUNDARY when averaged over a month exceeds 0.6 mrem to any organ.

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

2. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit a report to the Commission within 30 days, pursuant to Specification 6.9.3.g.

f. Gas Decay Tanks

The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 curies noble gases (considered as Xe-133).

1. The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per day when radioactive materials are being added to the tank and the reactor coolant system total activity exceeds the activity limit of Specification 3.1.4.

2. With the quantity of radioactive material in any gas decay 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.

g. Explosive Gas Mixture

The concentration of oxygen in the GAS DECAY TANK SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

1. The concentration of hydrogen and oxygen in the in-service gas decay tank shall be determined to be within the above limits by continuously* monitoring the waste gases in the in-service gas decay tank with a hydrogen and oxygen monitor required OPERABLE by Table 3.9-4.

2. With the concentration of oxygen in the in-service gas decay tank 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.

3. With the concentration of oxygen in the in-service gas decay tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume:

   (1) Immediately suspend all additions of waste gases to the affected gas decay tank, and

   (2) Either release the affected gas decay tank without delay in compliance with Specification 3.9.2.a, or commence reduction of oxygen concentration to less than or equal to 2% by volume without delay.

h. Total Dose

The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC from all uranium fuel cycle sources shall be limited to $\leq$ 25 mrem to the total body or any organ except the thyroid, which shall be limited to $\leq$ 75 mrem.

1. Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.9.1.b, 3.9.2.b and 3.9.2.c, and in accordance with the ODCM.

2. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.9.1.b, 3.9.2.b, or 3.9.2.c, calculations should be made to determine whether the above limits have been exceeded. If such is the case, prepare and submit a report to the Commission within 30 days, pursuant to Specification 6.9.3.h.

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* Until modifications to the gas analyzing equipment are complete, monitoring will consist of grab sampling and analysis at the following frequencies:

a) Once per 8 hours during degassing operations

b) Once per day during other operations.

TABLE 3.9-3

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING[d] FREQUENCY	MINIMUM[d] ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[a] (μCi/cc)
A. Gas Decay Tank (Batch)	PR Each Tank Grab Sample	PR Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment Purge (Batch)	PR[g] Grab Sample	PR[g]	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
C. Condenser Air Ejectors	M[g] Grab Sample	M[g] Gas Sample	Principal Gamma Emitters[b]	1×10^{-4}
			H-3	1×10^{-6}
	Continuous[c] *	4/M * Charcoal Sample	I-131	1×10^{-12}
	Continuous[c] *	4/M * Particulate Sample	Principal Gamma Emitters[b]	1×10^{-11}
D. Plant Vent and #3 Spent Fuel Pit Building	M[g] Grab Sample	M[g] Gas Sample	Principal Gamma Emitters[b]	1×10^{-4}
	M[e,f] Grab Sample	M	H-3	1×10^{-6}
	Continuous[c]	4/M[h] Charcoal Sample	I-131	1×10^{-12}
	Continuous[c]	4/M[h] Particulate Sample	Principal Gamma Emitters[b]	1×10^{-11}
	Continuous[c]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[c]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
E. Plant Vent	Continuous[c]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

* Dependent on operability of Condenser Air Ejector SPING-4 monitor. In the interim, grab samples will be taken and analyzed at least weekly when primary to secondary leakage is detected.

TABLE 3.9-3 (Cont.)

TABLE NOTATION

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

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

Where:

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

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

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

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 3.9-3 (Cont.)

TABLE NOTATION

b. The principal gamma emitters for which the LLD limit specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for noble gas emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99m, Cs-134, Cs-137, Ce-141, Ce-144, and I-131 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When a radionuclide's calculated LLD is greater than its listed LLD limit, the calculated LLD should be assigned as the activity of the radionuclide; or, the activity of the radionuclide should be calculated using measured ratios with those radionuclides which are routinely identified and measured.

c. 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 3.9.2.a, 3.9.2.b, and 3.9.2.c.

d. Frequency designations are defined in Table 4.1-1, sheet 4.

e. Tritium grab samples shall be taken at least once per day when the refueling canal is flooded.

f. Tritium grab samples shall be taken at least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

g. Sampling and analysis shall also be performed following shutdown, startup or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

h. Sample collection media shall be changed at least 4 times a month at intervals of no greater than 9 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 day for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3. When samples collected daily are analyzed, the corresponding LLDs may be increased by a factor of 10.

TABLE 3.9-4

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. Plant Vent			
a. Noble Gas Activity Monitor	(1)	*	4
b. Iodine Sampler Cartridge	(1)	*	3
c. Particulate Sampler Filter	(1)	*	3
d. Sampler Flow Rate Measuring Device	(1)	*	1
2. #3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	(1)	*	4
b. Iodine Sampler Cartridge	(1)	*	3
c. Particulate Sampler Filter	(1)	*	3
d. Sampler Flow Rate Measuring Device	(1)	*	1
3. Condenser Air Ejector Vent***			
a. Noble Gas Activity Monitor	(1)	*	2
b. Iodine Sampler Cartridge	(1) NOTE 1	**	3
c. Particulate Sampler Filter	(1) NOTE 1	**	3
d. Effluent System Flow Rate Measuring Device	(1)	**	1

TABLE 3.9-4

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. GAS DECAY TANK SYSTEM Explosive Gas Monitoring System			
a. Hydrogen and Oxygen	(1)	****	5(NOTE 2)

* At all times

** Daily when primary to secondary leakage is detected (as indicated by the condenser air ejector noble gas activity monitor)

*** N.A. during cold or refueling shutdowns

**** During GAS DECAY TANK SYSTEM operation

NOTE 1 Dependent on operability of condenser air ejector SPING-4 monitor. In the interim, the minimum channels OPERABLE requirement will be met by grab sampling and analysis.

NOTE 2 Until modifications to the gas analyzing equipment are complete, the following action will be taken:

With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately initiate corrective actions to restore the channel to OPERABLE status.

TABLE 3.9-4 (Cont.)

ACTION 1 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 every 8 hours.

ACTION 2 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 collected every 12 hours and analyzed within 24 hours for noble gas activity when primary to secondary leakage is occurring.

ACTION 3 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided samples are continuously collected with auxiliary sampling equipment and analyzed at least four times a month.

ACTION 4 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases, excluding gas decay tank releases, via this pathway may continue provided grab samples are collected at least once per 12 hours and these samples are analyzed for activity within 24 hours.

For Gas Decay Tank Releases:

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

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

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup (one performs, one verifies);

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 5* With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the GAS DECAY TANK SYSTEM may continue for up to 30 days provided that grab samples are collected and analyzed for hydrogen and oxygen concentration at least once per day.

* See NOTE 2 in Table 3.9-4

3. RADIOACTIVE BEAD RESINS

a. Radioactive bead resins shall be dewatered, as appropriate, in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

1. Prior to disposal, each container of radioactive bead resins shall be tested for free standing liquids in accordance with the PROCESS CONTROL PROGRAM to assure that it meets shipping, transportation and disposal site requirements.

2. With dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately dewatered bead resin and correct the PROCESS CONTROL PROGRAM, the procedures and/or the dewatering system as necessary to prevent recurrence.

3. With dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) if the dewatered bead resin has not already been shipped for disposal, verify each container to ensure that it meets burial ground and shipping requirements and (2) take appropriate administrative action to prevent recurrence.

TABLE 4.1-1 SHEET 2

Channel Description	Check	Calibrate	Test	Remarks
10. Rod Position Bank Counters	S†	N.A.	N.A.	With analog Rod Position
11. Steam Generator Level	S†	R	M†	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14. Boric Acid Tank Level	W	R	N.A.	
15. Refueling Water Storage Tank Level	W†	R	N.A.	
16. Volume Control Tank Level	N.A.	R	N.A.	
17A. Containment Pressure	D††	R	M††	Wide Range
17B. Containment Pressure	D††	R	M††	Narrow Range
18A. Process Radiation ***	D	N.A.	M	
18B. Area Radiation	D	A	M	
19. Boric Acid Control	N.A.	N.A.	R	
20. Containment Sump Level	N.A.	R	N.A.	
21. Accumulator Level and Pressure	S†	R	N.A.	
22. Steam Line Pressure	S†	R	M†	

TABLE 4.1-1 SHEET 3

	Channel Description	Check	Calibrate	Test	Remarks
23.	Logic Channels	N.A.	N.A.	M†	
24.	Emergency Portable Survey Instruments	N.A.	A	M	
25.	Seismograph	N.A.	N.A.	Q	Make trace, Test battery (change semi-annually)
26.	Auxiliary Feedwater Flow Rate	M†	R	N.A.	
27.	RCS Subcooling Margin Monitor	M†	R	N.A.	
28.	PORV Position Indicator (Primary Detector)	M†	N.A.	R	Check consists of monitoring indicated position and verifying by observation of related parameters
29.	PORV Block Valve Position Indicator	M†	N.A.	R	
30.	Safety Valve Position Indicator	M†	R	N.A.	
31.	Loss of Voltage (both 4kv busses)	N.A.	N.A.	R	For AFW actuation at power only
32.	Trip of Both Main Feedwater Pump Breakers	N.A.	N.A.	R	For AFW actuation at power only

TABLE 4.1-1 SHEET 4

* Using moveable in-core detector system
** Frequency only
*** Applies to non-effluent monitors only. For effluent monitors, see Tables 4.1-3 and 4.1-4.
PR - Prior to each release
S - Each Shift
D - Daily
W - Weekly
4/M - At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year
B/W - Every Two Weeks
M - Monthly
Q - Quarterly
P - Prior to each startup if not done previous week
R - Each Refueling Shutdown
A - Annually
N.A. - Not applicable

† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to startup.

†† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to heatup above 200F.

TABLE 4.1-2 (Sheet 2 of 3)

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

11.	Reactor Coolant System Leakage	Evaluate	Daily	NA
12.	Diesel Fuel Supply	Fuel Inventory	Weekly	10
13.	Spent Fuel Pit	Boron Concentration	Prior to Refueling	NA
14.	Fire Protection Pump & Power Supply	Operable	Monthly	45
15.	Turbine Stop and Control Valves, Reheater Stop and Intercept Valves	Closure	Monthly*	45
16.	LP Turbine Rotor Inspector (w/o rotor disassembly)	V, MT, PT	Every 5 Years	6 Years
17.	Spent Fuel Cask Crane Interlocks	Functioning	Within 7 Days	7 days when crane is being used to maneuver spent fuel cask

TABLE 4.1-2 (Sheet 3 of 3)

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

18. Coolant Loops	During power operation, verify three (3) reactor Coolant Loops in operation and circulating reactor coolant.	Once every 12 hrs.	12 hrs.
	At shutdown with average coolant temperature $\geq$ 350 F, verify		
	a. One (1) reactor Coolant Loop in operation and circulating reactor coolant.	Once every 12 hrs.	12 hrs
	b. A second Coolant Loop operable.	Once every 7 days	7 days
	At shutdown (not refueling) with average coolant temperature < 350 F, verify		
	a. One (1) Coolant Loop is in operation and circulating reactor coolant.	Once every 12 hrs.	12 hrs.
	b. A second Coolant Loop operable.	Once every 7 days.	7 days
	At refueling shutdown, verify that one (1) residual heat removal Coolant Loop is in operation and circulating sufficient reactor coolant to maintain core outlet temperature below 160 F.	Once every 4 hrs.	4 hrs.

† N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed prior to heatup above 200 F.

* - N.A. during cold or refueling shutdowns, or at hot shutdown when all main steam isolation valves are shut. The specified tests, however, shall be performed within one surveillance period prior to starting the turbine.

TABLE 4.1-3 SHEET 1

MINIMUM FREQUENCY FOR SURVEILLANCE OF RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation	(3)	(3)	(3)
a. Liquid Radwaste Effluent Line	D,PR	R	Q(1)
b. Steam Generator Blowdown Effluent Line	D	R	Q(1)
2. Flow Rate Monitor			
a. Liquid Radwaste Effluent Line	D(2)	R	Q
b. Steam Generator Blowdown Effluent Line	D(2)	R	Q

TABLE 4.1-3 SHEET 2

TABLE NOTATION

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

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

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

(3) Frequency designations are defined in Table 4.1-1, Sheet 4.

TABLE 4.1-4 SHEET 1

MINIMUM FREQUENCY FOR SURVEILLANCE OF RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK (5)	CHANNEL CALIBRATION (5)	CHANNEL FUNCTIONAL TEST (5)
1. Plant Vent			
a. Noble Gas Activity Monitor	D(1)	R	Q(3)
b Iodine Sampler Cartridge	W	N/A	N/A
c. Particulate Sampler Filter	W	N/A	N/A
d. Sampler Flow Rate Measuring Device	D	R	R
2. #3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	D	R	N/A
b. Iodine Sampler Cartridge	W	N/A	N/A
c. Particulate Sampler Filter	W	N/A	N/A
d. Sampler Flow Rate Measuring Device	D	R	R
3. Condenser Air Ejector Vent			
a. Noble Gas Activity Monitor	D	R	Q(2)
b. Iodine Sampler Cartridge *	W	N/A	N/A
c. Particulate Sampler Filter *	W	N/A	N/A
d. Effluent System Flow Rate Measuring Device	D	R	R

TABLE 4.1-4 SHEET 2

MINIMUM FREQUENCY FOR SURVEILLANCE OF RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. Containment Purge System			
a. Noble Gas Activity Monitor - Providing alarm and automatic termination of release.	D	A	Q(3)
b. Particulate Activity Monitor - Providing alarm and automatic termination of release.	D	A	Q(3)
c. Sampler Flow Rate Monitor	D	A	A
5. GAS DECAY TANK SYSTEM Explosive Gas Monitoring System			
a. Hydrogen and Oxygen Monitor	D	Q(4)	M

* Dependent on operability of condenser air ejector SPING-4 monitor.

TABLE NOTATION

(1) A CHANNEL CHECK for the plant vent noble gas activity monitor is performed prior to making a release through the GAS DECAY TANK SYSTEM release pathway.

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

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

(3) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway (the GAS DECAY TANK SYSTEM release pathway or the Containment Purge System Pathway, whichever is applicable) and control room alarm annunciation occurs if the following conditions exist:

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

(4) The CHANNEL CALIBRATION shall include the use of standard gases containing a known concentration of hydrogen and oxygen.

(5) Frequency designations are defined in Table 4.1-1, Sheet 4.

4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

Applicability: Applies to routine monitoring of plant environs.

Objective: To provide representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure.

Specifications:

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 4.12-2 and the detection capabilities required by Table 4.12-3.

   a. With the radiological environmental monitoring program not being conducted as specified in Table 4.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.4.b, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

   b. With the confirmed* 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, prepare and submit a report to the Commission within 30 days, pursuant to Specification 6.9.3.i. 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 dose to a MEMBER OF THE PUBLIC is equal to or greater than the limits of Specifications 3.9.1.b, 3.9.2.b and 3.9.2.c. 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.

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* A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis, but in any case within 30 days.

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. Pursuant to Specification 6.9.4.a, 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).

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.

   a. With a land use census identifying a location(s) that yield(s) a calculated dose or dose commitment greater than the values currently being calculated in Specification 3.9.2.c, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.4.a.

   b. With a land use census identifying a location(s) that yield(s) a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 4.12.1, 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. Pursuant to Specification 6.9.4.a, 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).

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* Broad leaf vegetation sampling 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 4.12-1.4b shall be followed, including analysis of control samples.

c. The land use census shall be conducted during the growing season at least once per year 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 6.9.4.b.

3. Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission. This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

   a. With the analyses not being performed as required above, report the corrective actions to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.4.b.

   b. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.4.b.

TABLE 4.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM [a]

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations [b,c]	Sampling and Collection Frequency [d]	Type and Frequency of Analysis [d]
1. DIRECT RADIATION [e]	21 Monitoring Locations	Continuous monitoring with sample collection quarterly [f]	Gamma exposure rate - quarterly
2. AIRBORNE			
Radioiodine and Particulates	5 Locations	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	Radioiodine Filter: I-131 analysis weekly Particulate Filter: Gross beta radioactivity analysis ≥ 24 hours following a filter change [g] Gamma isotopic [h] analysis of composite [g] (by location) quarterly
3. WATERBORNE [k]			
a. Surface [h]	3 Locations [i]	Monthly	Gamma isotopic [h] & tritium analyses monthly
b. Sediment from shoreline	3 Locations	Semiannually	Gamma isotopic [h] analysis semiannually

TABLE 4.12-1 (cont.)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM [a]

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations [b,c]	Sampling and Collection Frequency [d]	Type and Frequency of Analysis [d]
4. INGESTION			
a. Fish and Invertebrates			
1. Crustacea	2 Locations	Semiannually	Gamma isotopic [h] analyses semianually
2. Fish	2 Locations	Semiannually	Gamma isotopic [h] analyses semiannually
b. Food Products			
1. Broad leaf vegetation	3 Locations [1]	Monthly when available	Gamma isotopic [h] and I-131 analyses monthly

TABLE 4.12-1 (Cont.)

TABLE NOTATION

a. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, corrective action shall be taken 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 6.9.4.b.

b. Specific parameters of distance and direction sector from the centerline of the plant vent stack, and additional description where pertinent, shall be provided for each sample location in Table 4.12-1 in a Table and figure(s) in the ODCM.

c. 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.

d. The following definition of frequencies shall apply to Table 4.12-1 only:

Weekly - Not less than once per calendar week. A maximum interval of 11 days is allowed between the collection of any two consecutive samples.

Semi-Monthly - Not less than 2 times per calendar month with an interval of not less than 7 days between sample collections. A maximum interval of 24 days is allowed between collection of any two consecutive samples.

Monthly - Not less than once per calendar month with an interval of not less than 10 days between sample collections.

Quarterly - Not less than once per calendar quarter.

Semiannually - One sample each between calendar dates (January 1 -June 30) and (July 1 - December 31). An interval of not less than 30 days will be provided between sample collections.

The frequency of analyses is to be consistent with the sample collection frequency.

e. 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 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.

TABLE 4.12-1 (Cont.)

TABLE NOTATION

f. Refers to normal collection frequency. More frequent sample collection is permitted when conditions warrant it.

g. Airborne particulate sample filters are analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. In addition to the requirement for a gamma isotopic on a composite sample, a gamma isotopic is also required for each sample having a gross beta radioactivity which is > 1.0 pCi/m^3 and which is also > 10 times that of the most recent control sample.

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

i. Off-shore grab samples.

k. Discharges from the Turkey Point Plant do not influence drinking water or ground water pathways.

l. Samples of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q, and one sample of similar broad leaf vegetation at an available location 15-30 km distant in the least prevalent wind direction based upon historical data in the ODCM.

TABLE 4.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
	Reporting Levels				
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	

* Since no drinking water pathway exists, a value of 30,000 pCi/ℓ is used. For drinking water samples, a value of 20,000 pCi/ℓ is used. This is 40 CFR Part 141 value.

** Applies to drinking water

*** An equilibrium mixture of the parent and daughter isotopes which corresponds to the reporting value of the parent isotope.

TABLE 4.12-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS [a]

Lower Limit of Detection (LLD) [b,c]

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	3,000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-Nb-95	15[e]					
I-131	1[d]	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[e]			15[e]		

* Since no drinking water pathway exists, a value of 3,000 pCi/ℓ is used. For drinking water samples, a value of 2,000 pCi/ℓ is used.

TABLE 4.12-3

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 6.9.4.b.

b. Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

c. 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:

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

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

TABLE 4.12-3 (Cont.)

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. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.4.b.

d. LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

e. An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/ℓ of the parent isotope.

5.0 DESIGN FEATURES

5.1 SITE

The Turkey Point Plant Units 3 and 4 are located on the west shore of Biscayne Bay adjacent to existing oil and gas fired steam generating Units 1 and 2 (432 MWe each), owned and operated by Florida Power and Light Company. The site is approximately 25 miles south of Miami, eight miles east of Florida City and nine miles southeast of Homestead. The minimum distance from either reactor to the site exclusion boundary as defined in 10 CFR 100.3 exceeds 4164 feet.

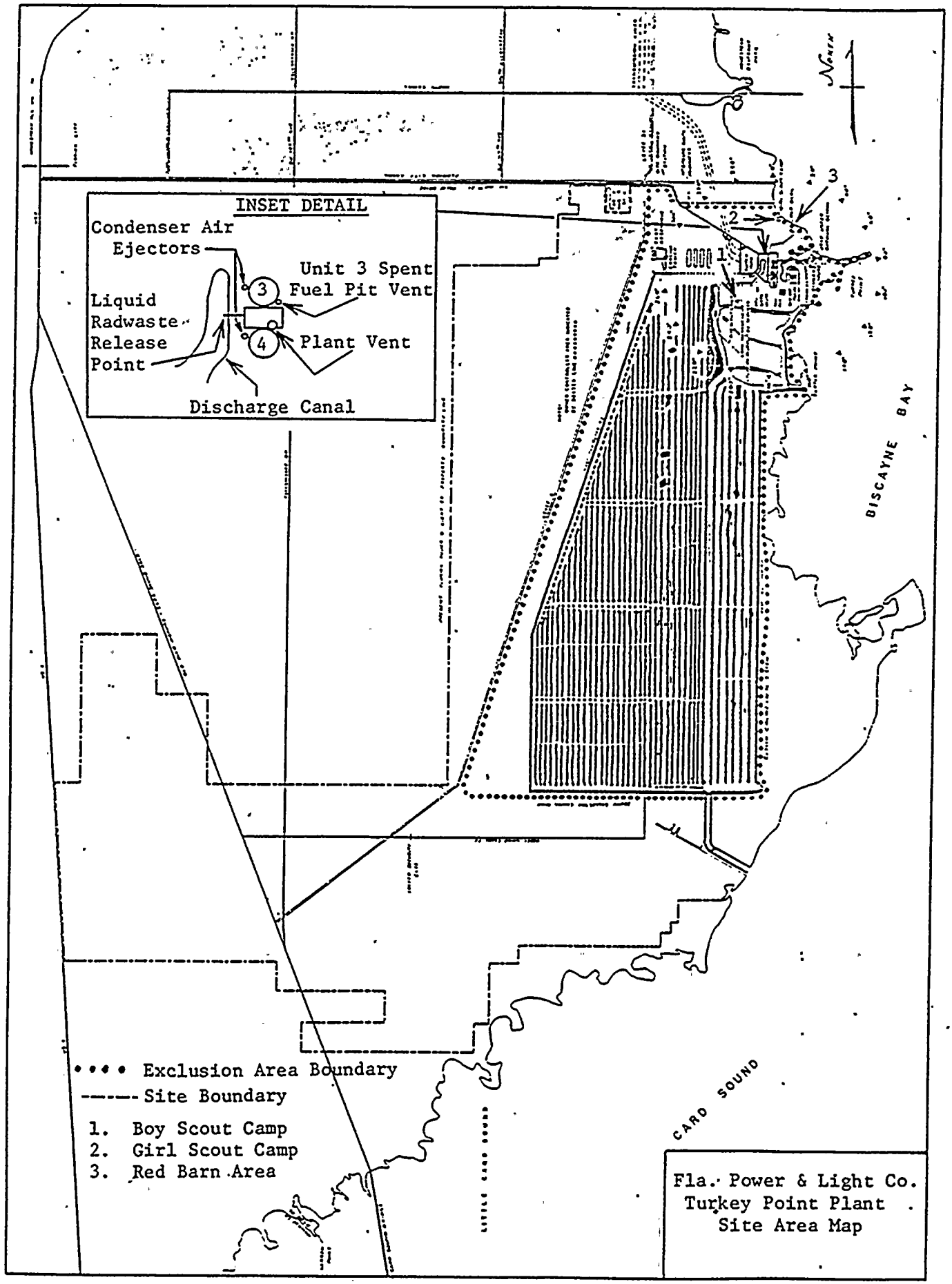
1. The exclusion area is shown on Figure 5.1-1.

   a. The following are UNRESTRICTED AREAS within the SITE BOUNDARY accessible to MEMBERS OF THE PUBLIC.

      1. Boy Scout Camp

      2. Girl Scout Camp

      3. Red Barn Area


INSET DETAIL
Condenser Air Ejectors
Unit 3 Spent Fuel Pit Vent
Liquid Radwaste Release Point
Plant Vent
Discharge Canal
3
4
North
1
2
3
BISCAYNE BAY
CARD SOUND
Exclusion Area Boundary
Site Boundary
1. Boy Scout Camp
2. Girl Scout Camp
3. Red Barn Area
Fla. Power & Light Co.
Turkey Point Plant
Site Area Map

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

g. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Company Nuclear Review Board.

h. Review of the Plant Security Plan and implementing procedures and submitting recommended changes to the Chairman of the Company Nuclear Review Board.

i. Review of the Emergency Plan and implementing procedures and submitting recommended changes to the Chairman of the Company Nuclear Review Board.

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

6.5.1.7 Authority

The Plant Nuclear Safety Committee shall:

a. Recommend to the Plant Manager - Nuclear written approval or disapproval (in minutes of PNSC meeting) of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing (in minutes of PNSC meetings) 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 immediate written notification to the Vice President of Nuclear Energy and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Manager - Nuclear; however, the Plant Manager - Nuclear shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

6.5.1.8 Records

The Plant Nuclear Safety Committee shall maintain written minutes of each meeting and copies shall be provided to the Vice President - Nuclear Energy and Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

6.5.2.1 Function

The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations.

b. Nuclear engineering.

c. Chemistry and radiochemistry.

d. Metallurgy.

e. Instrumentation and control.

f. Radiological safety.

g. Mechanical and electrical engineering.

h. Quality assurance practices.

6.5.2.2 Composition

The CNRB shall be composed of the:

h. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified licensee personnel or an outside fire protection firm.

i. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than three (3) years.

j. The radiological environmental monitoring program and the results thereof, at least once per year.

k. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once every two years.

l. The PROCESS CONTROL PROGRAM and implementing procedures for dewatering of radioactive bead resin at least once every two years.

m. The performance of activities required by the Quality Control Program to meet the criteria of Regulatory Guide 1.21, Revision 1 June 1974 and Regulatory Guide 4.1, Revision 1, April 1975, at least once per year.

n. Any other area of facility operation considered appropriate by the CNRB or the Executive Vice President.

6.5.2.9 AUTHORITY

The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Section 6.5.2.7 and 6.5.2.8.

6.5.2.10 RECORDS

Records of CNRB activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each CNRB meeting shall be prepared, approved and forwarded to the Executive Vice President within fourteen days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 e, f, g and h above, shall be prepared, approved and forwarded to the Executive Vice President within fourteen days following completion of the review.

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

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken in the event of a REPORTABLE OCCURRENCE:

a. The REPORTABLE OCCURRENCE shall be reported to the Commission pursuant to the requirements of Section 6.9.

b. A Reportable Occurrence Report shall be prepared. The report shall be reviewed by the Plant Nuclear Safety Committee.

c. The Reportable Occurrence Report shall be submitted to the CNRB, the Vice President of Nuclear Energy and the Commission within the time alloted in Section 6.9.

6.7 SAFETY LIMIT VIOLATION

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

a. The provisions of 10 CFR 50.36 (c) (1) (i) shall be complied with immediately.

b. The Safety Limit violation shall be reported immediately to the Commission, the Vice President of Nuclear Energy and to the CNRB.

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

d. The Safety Limit Violation Report shall be submitted to the CNRB, the Vice President of Nuclear Energy and the Commission within ten (10) days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures and administrative policies shall be established, implemented and maintained that meet or exceed: the requirements and recommendations of Section 5.1 and 5.3 of ANSI N18.7-1972, Appendix "A" of USNRC Regulatory Guide 1.33, PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, Quality Control Program for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974, Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975, and the Facility Fire Protection Program except as provided in 6.8.2 and 6.8.3 below.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, except the Quality Control Program for environmental monitoring, shall be reviewed by the PNSC and approved by the Plant Manager -Nuclear prior to implementation and periodically as provided by procedure.

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

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

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operators License on the unit affected.

c. The change is documented, reviewed by the PNSC and approved by the Plant Manager - Nuclear within fourteen days of implementation.

6.9.3 SPECIAL REPORTS

Special reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate.

Twenty copies of the following reports should be sent to the Director, Nuclear Reactor Regulation.

a. In-service inspection, reference 4.2

b. Tendon surveillance, reference 4.4

c. Fire protection systems, reference 3.14.

d. Peaking Factor Limit Report - The W(Z) function(s) for Base-Load Operation corresponding to a ±2% band about the target flux difference and/or a ±3% band about the target flux difference, the Load-Follow function $F_Z(Z)$ and the augmented surveillance turnon power fraction, P_T, shall be provided to the Director, Nuclear Reactor Regulations, Attention Chief of the Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 at least 60 days prior to cycle initial criticality, whenever P_T is < 1.0. In the event, the option of Baseload Operation (as defined in Section 3.2.6.a [3]) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_Z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the values would be submitted 60 days prior to the date the values would become effective unless otherwise approved by the Commission.

e. With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding the limits of Technical Specifications 3.9.1.b, 3.9.2.b, or 3.9.2.c, submit a report which 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 limits of 3.9.1.b, 3.9.2.b, or 3.9.2.c.

f. With untreated radioactive liquid effluents exceeding the limits of 3.9.1.d pursuant to Specification 3.9.1.d.3, submit a report which includes the following information:

(1) Identification of the inoperable equipment or subsystems and the reason for inoperability,

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

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

g. With untreated gaseous effluents exceeding the limits of 3.9.2.e pursuant to Specification 3.9.2.e.3, submit a report which includes the following information:

(1) Identification of the inoperable equipment or subsystems and the reason for inoperability,

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

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

h. With the annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC from all uranium fuel cycle sources exceeding the limits of Technical Specification 3.9.2.h, submit a report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.9.2.h and includes the schedule for achieving conformance with those limits. This 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 limits of Specification 3.9.2.h and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the 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.

i. With the measured levels of radioactivity in environmental samples as a result of plant effluents pursuant to Specification 4.12.1.b, submit a 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 dose to a MEMBER OF THE PUBLIC is less than the limits of Specifications 3.9.1.b, 3.9.2.b and 3.9.2.c.

6.9.4 UNIQUE REPORTING REQUIREMENTS

6.9.4.a SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

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 Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents 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 to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year.

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

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

The Radioactive Effluent Release Report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. Approximate and conservative approximate methods may be used in lieu of actual meteorological measurements. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases for the previous calendar year. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, March 1976.

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

a. Volume,

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

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

d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms)

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

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

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

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the OFFSITE DOSE CALCULATION MANUAL (ODCM) and the PROCESS CONTROL PROGRAM (PCP) as well as a listing of new locations for dose calculations and/or environmental monitoring:

a. Necessitated by the unavailability of environmental samples, pursuant to Specification 4.12.1.c; report shall also include the causes for unavailability of samples

b. Identified by the land use census pursuant to Specification 4.12.2.

6.9.4.b ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

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 Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and information based on trend analysis of the results of the radiological environmental surveillance activities for the report period, including a comparison, as appropriate, with preoperational studies, with operational controls and with previous environmental surveillance reports and an assessment of the observed impact of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 4.12.2.

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

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of the plant vent stack; the results of the Interlaboratory Comparison Program required by Specification 4.12.3; 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.

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

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

6.17 PROCESS CONTROL PROGRAM (PCP)

6.17.1 The PCP shall be reviewed by PNSC prior to implementation.

6.17.2 Licensee initiated changes to the PCP:

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

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

   b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

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

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

6.18 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.18.1 The ODCM shall be reviewed by the PNSC prior to submittal to the Commission.

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

6.18.3 Licensee initiated changes to the ODCM:

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

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

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

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

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

B3.9 BASES FOR LIMITING CONDITIONS FOR OPERATION, RADIOACTIVE MATERIALS RELEASE

B3.9.1 LIQUID EFFLUENTS

B3.9.1.a CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR 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.

B3.9.1.b DOSE

This specification is provided to implement the objectives of Appendix I, 10 CFR Part 50. The LIMITING CONDITIONS FOR OPERATION implement the guides set forth in Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". The dose calculation methodology and parameters in the ODCM implement the requirements in 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", March, 1976 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.

B3.9.1.c RADIOACTIVE LIQUID EFFLUENT 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 in accordance with the methodology 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.

B3.9.1.d LIQUID WASTE TREATMENT

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

B3.9.2 GASEOUS EFFLUENTS

B3.9.2.a 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 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of the MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE

BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

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

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.

B3.9.2.b DOSE - NOBLE GASES

This specification is provided to implement the objectives of Appendix I, 10 CFR Part 50. The LIMITING CONDITIONS FOR OPERATION implement the guides set forth in Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable". The surveillance requirements implement the requirements in 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 calculation methodology and parameters 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", March, 1976 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 or real atmospheric conditions.

This specification applies to the release of gaseous effluents from each reactor 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.

B3.9.2.c DOSE - IODINE-131, IODINE-133, TRITIUM AND PARTICULATES

This specification is provided to implement the objectives of Appendix I, 10 CFR Part 50. The LIMITING CONDITIONS FOR OPERATION implement the guides set forth in Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable". The surveillance requirements implement the requirements in 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 methodology and parameters 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", March, 1976 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 at and beyond the SITE BOUNDARY based upon the historical average atmospheric conditions or real atmospheric conditions. The release rate specifications for Iodine-131, Iodine-133, Tritium and particulates with half lives greater than 8 days are dependent upon 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.

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.

B3.9.2.d RADIOACTIVE GASEOUS EFFLUENT 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 in accordance with the methodology 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 the concentrations of potentially explosive gas mixtures in the GAS DECAY TANK 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.

B3.9.2.e GASEOUS RADWASTE TREATMENT

The requirements 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 objectives 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 objectives set forth in Appendix I, 10 CFR Part 50, for gaseous effluents.

B3.9.2.f GAS DECAY TANKS

Restricting the quantity of radioactivity contained in each gas decay 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 Standard Review Plan 15.7.1, "Waste Gas System Failure".

B3.9.2.g EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the GAS DECAY TANK 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.

B3.9.2.h TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a report whenever the calculated doses from plant generated radioactive effluents and direct radiation

exceed 25 mrems to the total body or any organ, except the thyroid, or exceed 75 mrems to the thyroid. For sites containing up to four 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 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 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 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 3.9.1.a and 3.9.2.a. 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.

B3.9.3 RADIOACTIVE BEAD RESINS

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 involved 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 and mixing and curing times.

B4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

B4.12.1 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 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. Analyses 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 6.9.4.b.

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).

B4.12.2 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, aerial survey or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than $50 m^2$

provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/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.

B4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of 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.