ENCLOSURE 1

TVA-SQN-TS-60

PROPOSED TECHNICAL SPECIFICATIONS

SEQUOYAH NUCLEAR PLANT
UNITS 1 AND 2

OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

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

PURGE - PURGING

1.21 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 requried to purify the confinement.

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.30 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

1.34 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.

SITE BOUNDARY

1.35 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

UNRESTRICTED AREA

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

MEMBER(S) OF THE PUBLIC

1.37 MEMBERS OF THE PUBLIC shall include all individuals who are not occupationally associated with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category does not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During releases via these pathways.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE take the ACTION shown in table 3.3-12. Exert best effort to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability could not be corrected within 30 days.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in table 4.3-8.

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 30 - 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:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity gamma at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 12 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity gamma at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-13

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Exert best efforts to return the instrument to OPERABLE status with 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the operability could not be corrected within 30 days.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas disposal system operation.

*** During shield building exhaust system operation.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 8 hours and these samples are analyzed for noble gas gross activity within 24 hours.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of these waste gas disposal system may continue for up to 14 days provided grab samples are taken and analyzed either (1) every 4 hours during degassing operations of the reactor coolant system, or (2) daily during other operations.

ACTION 44 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided that within 4 hours after the channel has been declared inoperable samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released to unrestricted areas (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material release to unrestricted areas exceeding the above limits, without delay, restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d] 1. Waste Condensate Tanks (3)	P - Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
2. Cask Decontamination Tank	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
3. Laundry Tanks (2) 4. Chemical Drain Tank 5. Monitor Tank 6. Distillate Tanks (2) 7. Condensate Demineralizer Waste Evaporator Blowdown Tank (1)	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e] 1. Steam Generator[h] Blowdown 2. Turbine[h] Building Sump	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	D Grab Sample	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE 4.11-1 (Continued)
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu Ci/ml^a$)
C. Periodic Continuous[e,h]	Continuous[g]	W Composite[c]	Principal Gamma Emitters[i]	$5x10^{-7}$
1. Non-Reclaimable Waste Tank			I-131	$1x10^{-6}$
2. High Crud Tank (2)	M[g] Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
3. Neutralizer Tank	Continuous[g]	M Composite[c]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous[g]	Q Composite[c]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

TABLE NOTATION

a. The LLD is defined for the purpose of these specifications as the samllest 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 a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\ s_b}{E\ .\ V\ .\ 2.22x10^6\ .\ Y\ .\ \exp(-\lambda.\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above in 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 countes per minute),

E is the counting efficiency as counts per disintegration

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

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

Y is the fractional radiochemical yield (when applicable),

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

Δt from plant effluents is the elapsed time between midpoint of sample collection and time of counting (MIDPOINT)

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 an a posteriori (after the fact) limit for a particulate measurement.

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

c. Prior to analyses, all samples taken for the composite shall be throughly mixed in order for the composite sample to be representative of the effluent release.

d. 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, by a method described in the ODCM, to assure representative sampling.

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

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

g. Releases from these tanks are continuously composited during releases. With the composite sampler or the sampler flow monitor inoperable, the sampling frequency shall be changed to require representative batch samples from each tank to be released to be taken prior to release and manually composite for these analyses.

h. Applicable only during periods of primary to secondary leakage or the release of radioactivity as detected by the effluent radiation monitor provided the radiation monitor setpoint is at a LLD of 1×10^{-6} Ci/ml and allowing for background radiation during periods when primary to secondary leakage is not occurring.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas shall be limited from each reactor unit:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a Licensee Event Report (LER), prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141.*

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

SURVEILLANCE REQUIREMENTS

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

*Applicable only if drinking water supply is taken from the receiving water body within three miles downstream of the plant discharge.

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent to unrestricted areas (see Figure 5.1-1) would exceed 0.06 mrem to the total body* or 0.2 mrem to any organ* in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment for more than 31 days and in excess of the above limits, in lieu of an LER, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*Per reactor unit

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 The dose rate due to I-131, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

Table 4.11-2
Radioactive Gaseous Waste Monitoring
Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab	P Each Tank	Noble Gases[g]	1×10^{-4}
B. Containment				
1. Purge	Pi Each Purge Grab Sample	Di Each Purge	Noble Gas[g] H-3	1×10^{-4} 1×10^{-6}
2. Vent	Dj Each Day Grab Sample	Dj Each Day	Noble Gas[g] H-3	1×10^{-4} 1×10^{-6}
C. Noble Gases and Tritium	M Grab Sample	M	Noble Gases[g]	1×10^{-4}
1. Condenser Vacuum Exhaust[h]			H-3	1×10^{-6}
2. Auxiliary Building Exhaust[b,e]				
3. Service Building Exhaust				
4. Shield Building Exhaust[b,c,h]				
D. Iodine and Particulates				
1. Auxiliary Building Exhaust	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
2. Shield Building Exhaust	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitter[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
E. Noble Gases all Releases types as listed in C	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for the 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 a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above in 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 disintegration,

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

b. Sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 15% of rated thermal power within 1 hour unless (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3 and (2) the noble gas activity monitor shows that the effluent activity has not increased by more than a factor of 3 and the noble gas monitor reading is at least 50% of the monitor setpoint.

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 2 days following each shutdown from ≥15% RATED THERMAL POWER, startup to ≥15% RATED THERMAL POWER or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that the effluent activity has not increased more than a factor of 3 and the noble gas monitor reading is at least 50% of the monitor setpoint.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fule pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for noble gases and Mn-54, Fe-59, I-131, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate principal gamma emitters. This list does not mean that only these nuclides are to be detected and reported. Other gamma peaks that are measureable and identifiable, together with the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.12.

h. During releases via this exhaust system.

i. Purging - Applicable in MODES 1, 2, 3, and 4, the upper and lower compartments of the containment shall be sampled prior to PURGING. Prior to breaking containment integrety in MODE 5 or 6, the upper and lower compartments of the containment shall be sampled. The incore instrument room purge sample shall be obtained at the shield building exhaust between 20and 25 minutes following initiation of the incore instrument room purge.

j. Venting - Applicable in Modes 1, 2, 3, and 4; the containment will be vented to the containment annulus and then to the auxilary building via containment annulus fans. The lower containment compartment shall be sampled daily when venting is to occur to account for the radioactivity being discharged from the venting process.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: At all times.

ACTION

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

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

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

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from I-131, Tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the site boundary (see Figure 5.1-1) shall be limited to the following from each reactor unit.

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

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment for more than 31 days and in excess of the above limits, in lieu of a Licensee Event Report (LER), prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*These doses are per reactor unit.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in a waste gas holdup 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.

b. With the concentration of oxygen in a waste gas holdup tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, without delay suspend all additions of waste gases to the affected waste gas holdup tank and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by monitoring the waste gas additions to the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 50,000 curies of noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made to determine if the above limits have been violated. If such is the case, in lieu of an LER, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report, as defined in 10 CFR 20.405c, shall include an analysis which estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources (including all effluent) pathways and direct radiation) for a calendar year that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in table 3.12-1, in lieu of a LER, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of specifications of 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

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

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

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a licensee event

LIMITING CONDITION FOR OPERATION

report (LER) and pursuant to Specification 6.9.2, identify the cause(s) of the unavailability of samples and identify the new locations for obtaining replacement samples in the Annual Radiological Environmental Operating Report. A revised figure(s) and table(s) for the ODCM reflecting the new location(s) shall be included in in the next annual radiological effluent release report pursuantto specification 6.9.1.9.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of tables 3.12-1 and the detection capabilities required by table 4.2.1.

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is defined, for the 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 a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above in picocurie 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.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analysis will be performed in such a manner that the stated LLDs will be achieved under routine conditions.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the maximum value currently being calculated in Specification 4.11.2.3.1, in lieu of a licensee event report (LER), identify the new location(s) in the next Annual Radiological Environmental Operating Report pursuant to specification 6.9.1.7.

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than the highest calculated dose or dose commitment at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of a licensee event report (LER) identifying the new location(s) in the Annual Radiological Environmental Operating Report and include the revised ODCM table(s) and figure(s). The new location shall be added to the radiological environmental monitoring program within 30 days, if the owner consents. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (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.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per calendar year between the dates of April 1 and October 1 using the following techniques:

1. Within a 2-mile radius from the plant or within the 15-mrem/year isodose line, whichever is larger, enumeration by a door-to-door or equivalent counting technique.
2. With a 5-mile radius from the plant, enumeration by using appropriate techniques such as door-to-door survey, mail survey, telephone survey, serial survey, or information from local agricultural authorities or other reliable sources.

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

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas 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 exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 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.

3/4.11.1.2 DOSE

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

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

3/4.11.1.3 LIQUID WASTE TREATMENT

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

3/4.11.1.4 LIQUID HOLDUP TANKS

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

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

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

BASES

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.

3/4.11.2.2 DOSE, NOBLE GASES

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

3/4.11.2.3 DOSE - IODINE 131, TRITIUM, AND RADIONUCLIDES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodologies provided in NUREG/CR-1004, " A Statistical Ananlysis of Selected Parameters for Predicting Food Chain Transport and Internal Dose of Radionuclides", October, 1979 and Regulatory Guide 1.109, "Calculation of

BASES

Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for I-131, Tritium, and Radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man, beyond the site boundary. The pathways which were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.b of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

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

3/4.11.2.6 GAS 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".

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

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

The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. 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.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial and 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 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kgs/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, (1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kgs/square meter.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

6.1.2 The Chief, Radiological Services, shall be responsible for implementing the radiological environmental program and dose calculations and projections as described in the Offsite Dose Calculation Manual (ODCM). These responsibilities include performance of surveillance requirements listed in Table 6.1-1.

6.1.3 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Site Director, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2-1.

6.2.1.2 The offsite organization for the radiological enviromental monitoring program and dose calculations shall be as shown in Figure 6.2-3.

UNIT STAFF

6.2.2 The Unit organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

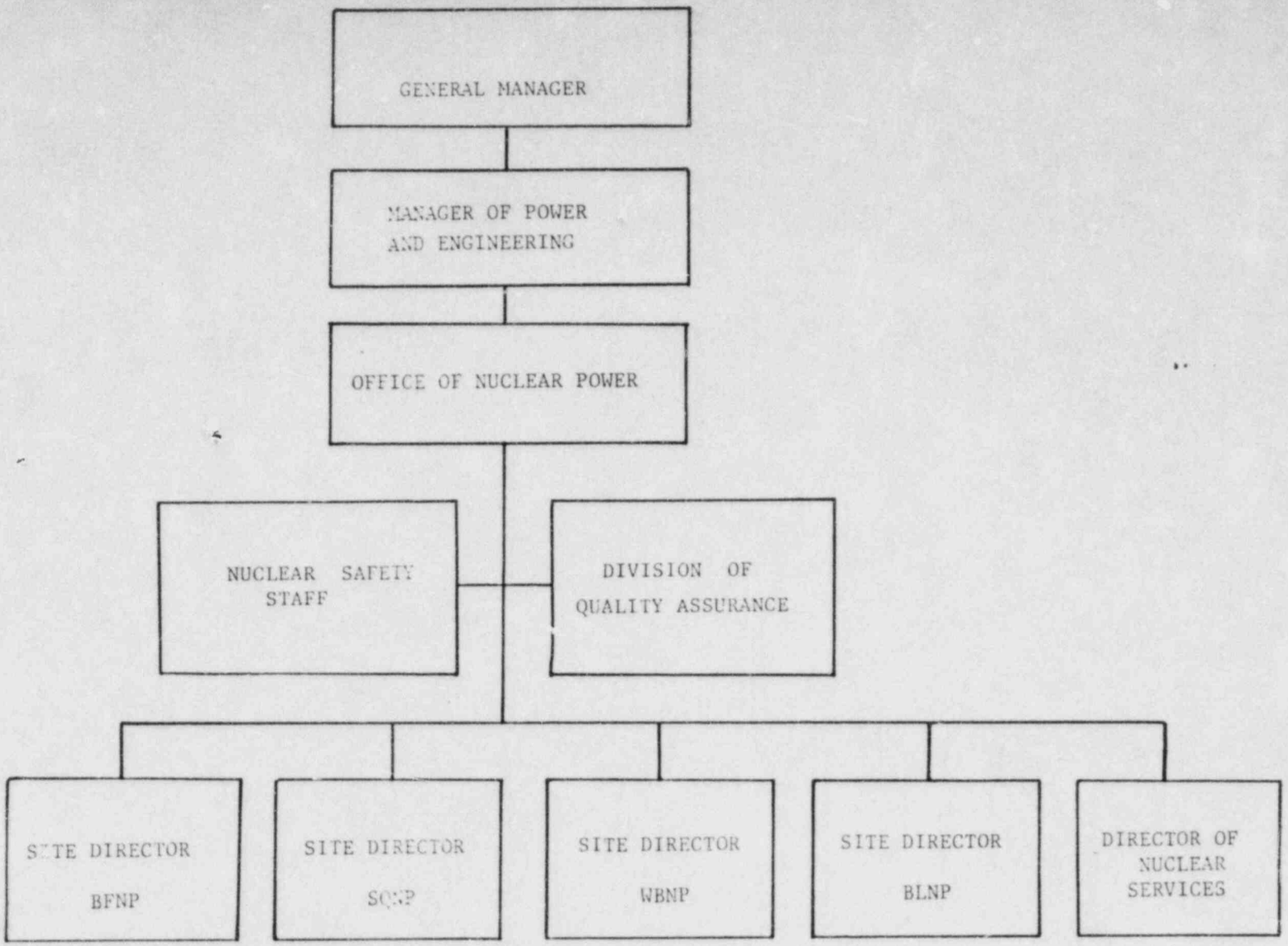
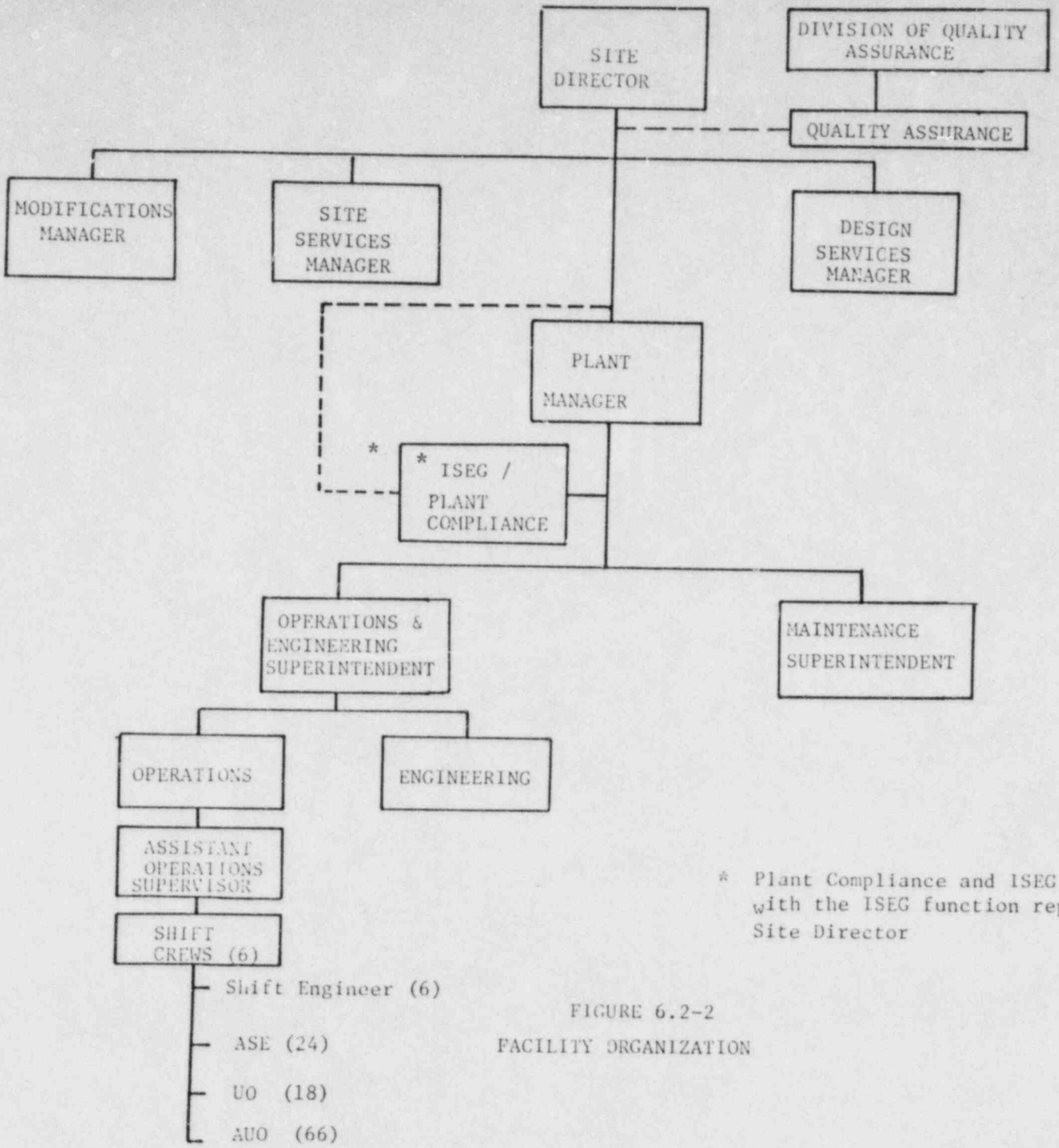


FIGURE 6.2-1

OFFSITE ORGANIZATION FOR FACILITY MANAGEMENT AND TECHNICAL SUPPORT



* Plant Compliance and ISEG are the same group with the ISEG function reporting to the Site Director

FIGURE 6.2-2

FACILITY ORGANIZATION

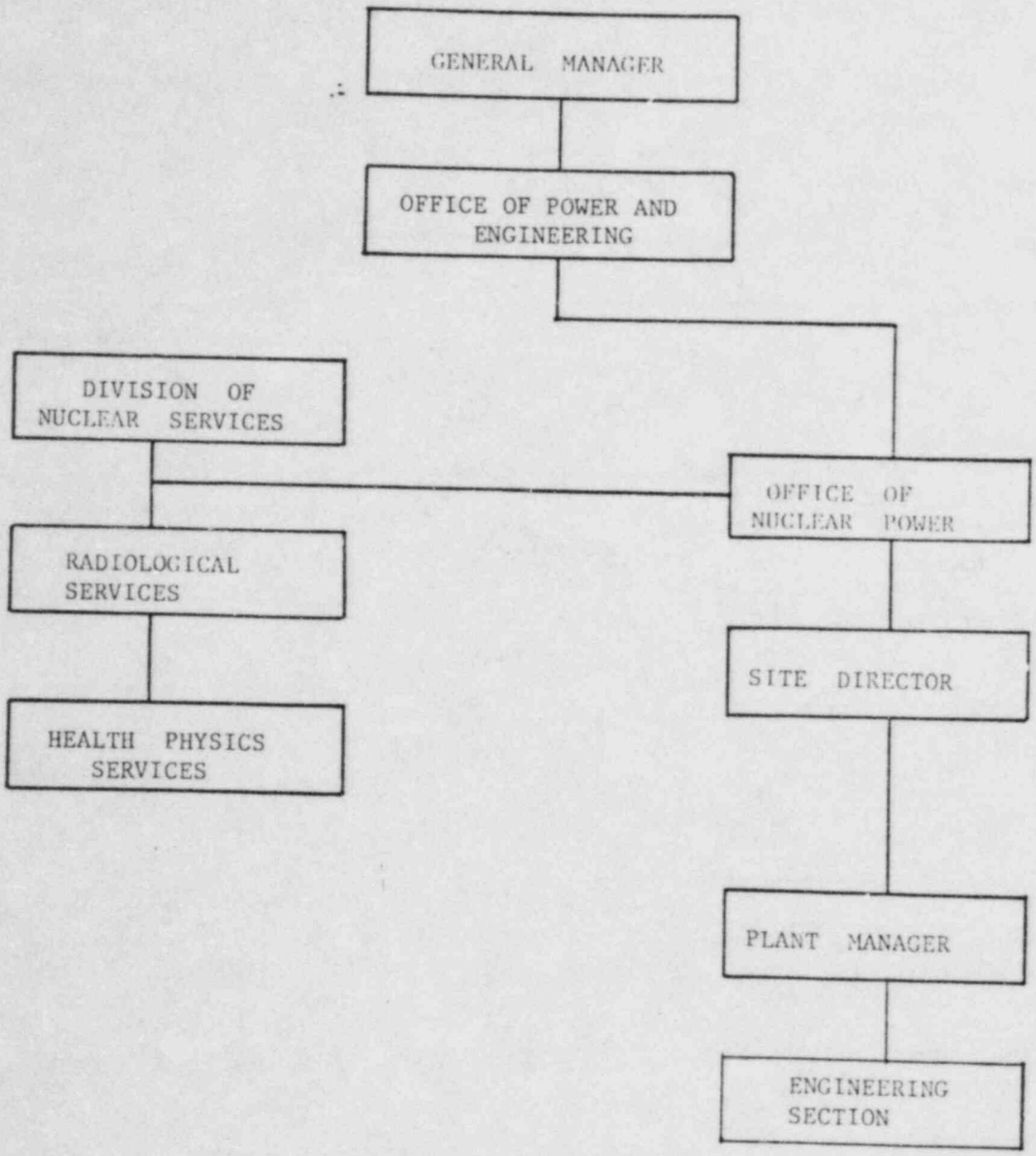


FIGURE 6.2-3

ORGANIZATION FOR MONITORING PROGRAM AND DOSE CALCULATIONS

TABLE 6.1-1

SURVEILLANCE REQUIREMENTS PERFORMED BY

: HEALTH PHYSICS SERVICES

4.11.1.2
4.11.1.3.1
4.11.2.1.1 (partial)
4.11.2.1.2 (partial)
4.11.2.2
4.11.2.3
4.11.2.4
4.11.4
4.12.1
4.12.2
4.12.3

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensing Event Reports and other sources which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five full time engineers located onsite.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Site Director.

6.2.4 SHIFT TECHNICAL ADVISOR (STA)

6.2.4.1 The STA shall serve in an advisory capacity to the shift supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of March 28, 1980 NRC letter to all licensees, except for the Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Operations and Engineering Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ISEG.

6.5 REVIEW AND AUDIT

6.5.0 The Manager of the Office of Nuclear Power is responsible for the safe operation of all TVA nuclear power plants. The functional organization for review and audit cognizance is shown on Figure 6.2-1.

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Plant Manager
Member:	Operations Supervisor or Assistant Operations Supvr.
Member:	Engineering Group Supervisor or Engr. Section Supvr.
Member:	Maintenance Supervisor
Member:	Operations/Engineering Superintendent or Maintenance Superintendent
Member:	Health Physicist
Member:	Supervisor Quality Assurance

ALTERNATES

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

MEETING FREQUENCY

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

QUORUM

6.5.1.5 The minimum quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8.1 and changes thereto, 2) all programs required by Specification 6.8.5, and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

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

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

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

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Site Director and to the Chief of the Nuclear Safety Staff.

f. Review all written reports requiring notification to the Commission.

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

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Chief, Nuclear Safety Staff.

i. Review of the Plant Physical Security Plan and implementing procedures and shall submit recommended changes to the Chief, Nuclear Safety Staff.

j. Review of the Site Radiological Emergency Plan and implementing procedures and shall submit recommended changes to the Chief, Nuclear Safety Staff.

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Director, Nuclear Power Division and to the Chief, Nuclear Safety Staff.

l. Review of changes to the radwaste treatment systems.

m. Review of meeting minutes of the Radiological Assessment Review Committee (RARC).

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

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

c. Provide written notification within 24 hours to the Site Director and the Chief, Nuclear Safety Staff of disagreement between the PORC and the Plant Manager ; however, the Plant Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Plant Manager and to the Chief, Nuclear Safety Staff.

6.5.2 NUCLEAR SAFETY Staff (NSS)

FUNCTION

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

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

RESPONSIBILITY

6.5.2.2 The NSS shall be responsible for the independent nuclear safety review program and cognizance of audits for all TVA nuclear plants.

QUALIFICATIONS

6.5.2.3 All NSS and other TVA personnel who have been designated as reviewers shall have an academic degree in engineering or a physical science field, or the equivalent; and in addition, shall have a minimum of five years of technical experience in one or more areas given in 6.5.2.1. The Chief, NSS shall meet the same qualifications as reviewers except that the minimum experience shall be six years.

CONSULTANTS

6.5.2.4 Consultants to the NSS shall be utilized to review or to provide expert advice as determined by the Chief, NSS.

MINIMUM REVIEW

6.5.2.5 A minimum of three reviewers shall review each of the subjects encompassed by sections 6.5.2.7 and 6.5.2.8.

REVIEW

6.5.2.6 The NSS shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

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

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All written reports requiring notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the PORC and the RARC.

AUDITS

6.5.2.7 Audits of unit activities shall be performed under the cognizance of the NSS. These audits shall encompass:

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

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

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

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

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

f. The Plant Physical Security Plan, the Safeguards Contingency Plan, and implementing procedures at least once per 12 months.

g. Any other area of unit operation considered appropriate by the NSS or the Manager of Nuclear Power.

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

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

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

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

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

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

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, December 1977 or Regulatory Guide 1.21 Rev.1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975, at least once per 12 months.

AUTHORITY

6.5.2.8 The NSS shall report to and advise the Manager of Nuclear Power areas of responsibility specified in Sections 6.5.2.6 and 6.5.2.7.

REPORTS

6.5.2.9 Reports of NSS activities shall be prepared, approved and distributed as indicated below:

a. Results of reviews and of cognizance of audits, including those encompassed by Sections 6.5.2.6 and 6.5.2.7 above, shall be approved by the Chief, NSS and forwarded to the Manager of Nuclear Power at least quarterly.

b. Audit reports encompassed by Section 6.5.2.7 above, shall be forwarded to the Manager of Nuclear Power and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.5.3 RADIOLOGICAL ASSESSMENT REVIEW COMMITTEE (RARC)

Function

6.5.3.1 The SQN RARC shall function to advise the Chief, Health Physics Services, and the Plant Manager on all matters related to radiological assessments involving dose calculations and projections and environmental monitoring.

Composition

6.5.3.2 The RARC shall be composed of the:

Chairman: Assessment Unit Supervisor, Health Physics Services
Member: Health Physicist, Gaseous, Health Physics Services
Member: Health Physicist, Liquid, Health Physics Services
Member: Meteorologist, Air Quality Branch
Member: Chemical Engineer, Engineering Section, SQN
Member: Office of Nuclear Power Representative

ALTERNATES

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

MEETING FREQUENCY

6.5.3.4 The SQN RARC shall meet at least once per six months or as requested by the SQN RARC Chairman or his/her designated alternate or a plant representative.

QUORUM

6.5.3.5 The minimum quorum of the SQN RARC necessary for the performance of the SQN RARC responsibility and authority provisions of these technical specifications shall consist of the Chairman or his designated alternate and 3 members (including alternates) as long as one is a plant representative.

RESPONSIBILITIES

6.5.3.6 The RARC shall be responsible for:

a. Review of changes to the OFFSITE DOSE CALCULATION MANUAL.

b. Review of all procedures required by Specification 6.8.4 and changes thereto.

c. Review for information purposes results of any audits, review or evaluation of the Quality Assurance Program for effluent and environmental monitoring and radiological assessments involving dose evaluations and projections.

d. Review of proposed changes to the technical specifications related to plant effluent radiological assessments involving dose calculations and projections and environmental radiological monitoring.

Authority

6.5.3.7 The RARC shall:

a. Recommend in writing to the Chief, Health Physics Services and to PORC approval or disapproval of items considered under 6.5.3.6 above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.3.6 constitutes an unreviewed safety question.

c. Provide timely written notification to the Site Director, and the Nuclear Safety Staff of unresolvable items associated with a and b above between the RARC and the Chief, Health Physics Services; however, the Chief, Health Physics Services, in coordination with the Plant Manager, shall have the responsibility for resolution of any such disagreement pursuant to 6.1.2 above.

Records

6.5.3.8 The SQN RARC shall maintain written minutes of each RARC meeting that at a minimum, document the results of all RARC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Site Director, PORC, and the Chief, Nuclear Safety Staff.

6.6 REPORTABLE EVENTS

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Sections 50.72 and 50.73 to 10 CFR Part 50 and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the report shall be submitted to the Chief, NSS and the Site Director.

6.7 SAFETY LIMIT VIOLATION

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

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

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Director and the Chief, NSS shall be notified within 24 hours.

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

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chief, NSS and the Site Director within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Plant Physical Security Plan implementation.

e. Site Radiological Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. Quality Assurance Program for effluent monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977 or Regulatory Guide 1.21 Rev.1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975.

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

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

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

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

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

6.8.4 Written procedures shall be established, implemented and maintained by the Health Physics Services covering the activities below:

a. OFFSITE DOSE CALCULATIONAL MANUAL implementation.

b. Quality Assurance Program for environmental monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977.

c. Surveillance requirements and environmental monitoring requirements shown in Table 6.1-1.

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

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

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

b. In-Plant Radiation Monitoring

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

(i) Training of personnel,

(ii) Procedures for monitoring, and

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

c. Secondary Water Chemistry

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

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

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

(iii) Identification of process sampling points,

ANNUAL REPORTS[1/]

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

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3/]

6.9.1.6 The radiological environmental operating report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an anlysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparision with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. and a listing of the new locations for dose calculations and/or environmental monitoring identified by the land use census. If harmfull effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problems and a planned course of action to alleviate the problem.

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

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

3/ 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 matrial from each unit.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[1/]

6.9.1.8 The semiannual radioactive effluent release report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

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

The annual radioactive effluent release report (Radiological Impact) to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same 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. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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

The annual radioactive effluent release report to be submitted after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed members of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The semiannual radioactive effluent release reports shall include the following information for each type ;of solid waste identified in Regulatory 1.21, Rev.1, Table 3, Part A, which is shipped offsite during the report period:

a. Total volume of containers,

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

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

d. Type of quantity (e.g., LSA, Type A, Type B, etc.)

The semiannual radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis, and shall include any changes to the PROCESS CONTROL PROGRAM (PCP) and the Offsite Dose Calculation Manual (ODCM) made during the reporting period. It shall include the type of solidification agent used, if applicable.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or Safety Valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PORC.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The W(z) function for normal operation shall be provided to the Director, Nuclear Reactor Regulation, 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. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.

Any information needed to suport W(z) will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

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

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

a. Records and logs of unit operation covering time interval at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.
c. All REPORTABLE EVENTS submitted to the Commission.
d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.
e. Records of changes made to the procedures required by Specification 6.8.1 and 6.8.4.
f. Records of radioactive shipments.
g. Records of sealed source and fission detector leak tests and results.
h. Records of annual physical inventory of all sealed source material of record.

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

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

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

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

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

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

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

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

i. Records of Quality Assurance activities required by the Operational Quality Assurance Manual.

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

k. Records of meetings of the PORC, RARC, and the NSS reports of the results of review and of cognizance of audits.

l. Records of analyses required by the radiological environmental monitoring program.

m. Records of secondary water sampling and water quality.

n. Records of the service life monitoring of all hydraulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b, including the maintenance performed to renew the service

o. Records for Environmental Qualific... on which are covered under the provisions of Paragraph 2.c.(12)(6) of License No. DPR-77.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

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

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

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

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special (Radiation) Work Permit.

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

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual 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,

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

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

c. documentation of the fact that the change has been reviewed and found acceptable by the PORC.

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the next annual radioactive release report pursuant to specification 6.9.1.9. This submittal shall contain:

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

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

c. documentation of the fact that the change has been reviewed and found acceptable by the RARC.

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

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

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

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewd by the PORC. The discussion of each change shall contain:

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

OFFSITE DOSE CALCULATION MANUAL

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

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

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

SHUTDOWN MARGIN

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

1.29 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

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

VENTILATION EXHAUST TREATMENT SYSTEM

1.33 A VENTILATION EXHAUST TREATEMENT 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 absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.34 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.

SITE BOUNDARY

1.35 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee (see figure 5.1-1).

UNRESTRICTED AREA

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

MEMBERS OF THE PUBLIC

1.37 MEMBERS OF THE PUBLIC shall include all individuals who are not occupationally associated with the plant. This category shall include non-employees of the licensee who are permitted to use portions of the site for recreational, occupational, or other purposes not associated with plant functions. This category does not include non-employees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During releases via these pathways.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE take the ACTION shown in table 3.3-12. Exert best effort to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the inoperability could not be corrected within 30 days.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in table 4.3-8.

TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 30 - 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:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity gamma at a limit of detection of at least 10^{-7} microcuries/gram:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram DOSE EQUIVALENT I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram DOSE EQUIVALENT I-131.

ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity gamma at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue provided the tank liquid level is estimated during all liquid additions to the tank.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in table 3.3-13

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in table 3.3-13. Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report why the operability could not be corrected within 30 days.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.13.b are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST at the frequencies shown in table 4.3-9.

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.
** During waste gas disposal system operation.
*** During shield building exhaust system operation.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the grab samples are taken at least once per 8 hours and these samples are analyzed for noble gas gross activity within 24 hours.

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this waste gas disposal system may continue for up to 14 days provided grab samples are taken and analyzed either (1) every 4 hours during degassing operations of the reactor coolant system or (2) daily during other operations.

ACTION 44 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continued provided that within 4 hours after the channel has been declared inoperable samples are continuously collected with auxiliary sampling equipment as required in table 4.11-2.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released to unrestricted areas (see figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released to unrestricted areas exceeding the above limits, without delay, restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d] 1. Waste Condensate Tanks (3) 2. Cask Decontamination Tank 3. Laundry Tanks (2) 4. Chemical Drain Tank 5. Monitor Tank 6. Distillate Tanks (2) 7. Condensate Demineralizer Waste Evaporator Blowdown Tank (1)	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e] 1. Steam Generator Blowdown[h] 2. Turbine Building Sump[h]	D Grab Sample	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	D Grab Sample	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	D Grab Sample	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) ($\mu Ci/ml^a$)
C. Periodic Continuous[e,h]	Continuous[g]	W Composite[c]	Principal Gamma Emitters[f]	$5x10^{-7}$
1. Non-Reclaimable Waste Tank			I-131	$1x10^{-6}$
2. High Crud Tank (2)	M[g] Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
3. Neutralizer Tank	Continuous[g]	M Composite[c]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	Continuous[g]	Q Composite[c]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$

TABLE NOTATION

a. The LLD is defined for the purpose of these specifications as the samllest 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 a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\ s_b}{E\ .\ V\ .\ 2.22x10^6\ .\ Y\ .\ \exp(-\lambda.\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above in 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 countes per minute),

E is the counting efficiency as counts per disintegration

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

TABLE 4.11-1 (Coninued)

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

Y is the fractional radiochemical yield (when applicable),

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

Δt from plant effluents is the elapsed time between midpoint of sample collection and time of counting (midpoint)

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 an a posteriori (after the fact) limit for a particulate measurement.

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

c. Prior to analyses, all samples taken for the composite shall be throughly mixed in order for the composite sample to be representative of the effluent release.

d. 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, by a method described in the ODCM, to assure representative sampling.

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

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

g. Releases from these tanks are continuously composited during releases. With the composite sampler or the sampler flow monitor inoperable, the sampling frequency shall be changed to require representative batch samples from each tank to be released to be taken prior to release and manually composite for these analyses.

h. Applicable only during periods of primary to secondary leakage or the release of radioactivity as detected by the effluent radiation monitor provided the radiation monitor setpoint is at a LLD of $\leq 1 \times 10^{-6}$ μCi/ml and allowing for background radiation during periods when primary to secondary leakage is not occurring.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted area (see figure 5.1-1) shall be limited from each reactor unit:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of a licensee event report (LER), prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141.*

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

SURVEILLANCE REQUIREMENTS

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

*Applicable only if drinking water supply is taken from the receiving water body within three miles downstream of the plant discharge.

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent to unrestricted areas (see figure 5.1-1) when averaged over 31 days, would exceed 0.06 mrem to the total body* or 0.2 mrem to any organ* in a 31 day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment for more than 31 days and in excess of the above limits, in lieu of a LER, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*Per reactor unit

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents to areas at or beyond the site boundary (unrestricted areas) (see figure 5.1-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.11.2.1.2 The dose rate due to I-131, tritium, and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in table 4.11-2.

Table 4.11-2

Radioactive Gaseous Waste Monitoring
Sampling and Analysis Program

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab	P Each Tank	Noble Gases[g]	1×10^{-4}
B. Containment	Pi	Di	Noble Gas[g]	1×10^{-4}
1. Purge	Each Purge Grab Sample	Each Purge	H-3	1×10^{-6}
2. Vent	Dj Each Day Grab Sample	Dj Each Day	Noble Gas[g] H-3	1×10^{-4} 1×10^{-6}
C. Noble Gases and Tritium	M Grab	M	Noble Gases[g]	1×10^{-4}
1. Condenser Vacuum Exhaust[h]	Sample		H-3	1×10^{-6}
2. Auxiliary Building Exhaust[b,e]				
3. Service Building Exhaust				
4. Shield Building Exhaust[b,c,h]				
D. Iodine and Particulates	Continuous[f]	W[d]	I-131	1×10^{-12}
1. Auxiliary Building Exhaust	Sampler	Charcoal Sample		
2. Shield Building Exhaust	Continuous[f]	W[d]	Principal Gamma	1×10^{-11}
	Sampler	Particulate Sample	Emitter[g] (I-131, Others)	
	Continuous[f]	M	Gross Alpha	1×10^{-11}
	Sampler	Composite Particulate Sample		
	Continutous[f]	O	Sr-89, Sr-90	1×10^{-11}
	Sampler	Composite Particulate Sample		
E. Noble Gases all Releases types as listed in C	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for the 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 a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above in 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 disintegration,

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

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

Y is the fractional radiochemical yield (when applicable),

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

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

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

b. Sampling and analysis shall also be performed following shutdown, startup, or a thermal power change exceeding 15% of rated thermal power within 1 hour unless (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3 and (2) the noble gas activity monitor shows that the effluent activity has not increased by more than a factor of 3 and the noble gas monitor reading is at least 50% of the monitor setpoint.

TABLE 4.11-2 (Continued)

TABLE NOTATION

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

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 2 days following each shutdown from ≥15% RATED THERMAL POWER, startup to ≥15% RATED THERMAL POWER or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that the effluent activity has not increased more than a factor of 3 and the noble gas monitor reading is at least 50% of the monitor setpoint.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fule pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for noble gases and Mn-54, Fe-59, I-131, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate principal gamma emitters. This list does not mean that only these nuclides are to be detected and reported. Other gamma peaks that are measureable and identifiable, together with the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.12.

h. During releases via this exhaust system.

i. Purging - Applicable in MODES 1, 2, 3, and 4, the upper and lower compartments of the containment shall be sampled prior to PURGING. Prior to breaking containment integrety in MODE 5 or 6, the upper and lower compartments of the containment shall be sampled. The incore instrument room purge sample shall be obtained at the shield building exhaust between 20and 25 minutes following initiation of the incore instrument room purge.

j. Venting - Applicable in Modes 1, 2, 3, and 4; the containment will be vented to the containment annulus and then to the auxilary building via containment annulus fans. The lower containment compartment shall be sampled daily when venting is to occur to account for the radioactivity being discharged from the venting process.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents from each reactor unit to areas at or beyond the site boundary (see figure 5.1-1) shall be limited to the following*:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of a licensee event report (LER), prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in complicance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gas shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

*Per reactor unit

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

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMEBER OF THE PUBLIC from I-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the site boundary (see figure 5.1-1) shall be limited to the following from each reactor unit.

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

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see figure 5.1-1), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation* and 0.4 mrad for beta radiation*. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases to areas at or beyond the site boundary (see figure 5.1-1) when averaged over 31 days would exceed 0.3 mrem to any organ.*

APPLICABILITY: At all times.

ACTION:

a. With gaseous waste being discharged without treatment for more than 31 days and in excess of the above limits, in lieu of a licensee event report (LER), prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

*These doses are per reactor unit.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in a waste gas holdup 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.

b. With the concentration of oxygen in a waste gas holdup tank greater than 4% by volume and the hydrogen concentration greater than 2% by volume, without delay suspend all additions of waste gases to the affected waste gas holdup tank and reduce the concentration of oxygen to less than or equal to 2% by volume without delay.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen and oxyge in the waste gas holdup system shall be determined to be within the above limits by monitoring the waste gas additions to the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by table 3.3-13 of Specification 3.3.3.10.

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 50,000 curies of noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, without delay suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the process control program not satisfied, suspend shipments of defectively processed, or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9.6 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

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3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, calculations should be made to determine if the above limits have been violated. If such is the case, in lieu of a LER, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report as defined in 10 CFR 20.405c shall include an analysis which estimates the radiation exposure (dose) to A MEMBER OF THE PUBLIC from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a calendar year period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b).

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

SURVEILLANCE REQUIREMENTS

4.11.4 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in table 3.12-1, in lieu of a LER, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of specifications of 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

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

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

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a licensee event

LIMITING CONDITION FOR OPERATION

report (LER) and pursuant to Specification 6.9.2, identify the cause(s) of the unavailability of samples and identify the new locations for obtaining replacement samples in the Annual Radiological Environmental Operating Report. A revised figure(s) and table(s) for the ODCM reflecting the new location(s) shall be included in in the next annual radiological effluent release report pursuantto specification 6.9.1.9.

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of tables 3.12-1 and the detection capabilities required by table 4.2.1.

TABLE 4.12-1 (Continued)

TABLE NOTATION

a. The LLD is defined for the 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 a 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above in picocurie 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.

It should be recognized that the LLD is defined as a a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analysis will be performed in such a manner that the stated LLDs will be achieved under routine conditions.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the maximum value currently being calculated in Specification 4.11.2.3.1, in lieu of a licensee event report (LER) identify the new location(s) in the next Annual Radiological Environmental Operating Report pursuant to specification 6.9.1.7.

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than the highest calculated dose or dose commitment at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of a licensee event report (LER) identifying the new location(s) in the Annual Radiological Environmental Operating Report and include the revised ODCM table(s) and figure(s). The new location shall be added to the radiological environmental monitoring program within 30 days, if the owner consents. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (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.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per calendar year between the dates of April 1 and October 1 using the following techniques:

1. Within a 2-mile radius from the plant or within the 15-mrem/year isodose line, whichever is larger, enumeration by a door-to-door or equivalent counting technique.
2. With a 5-mile radius from the plant, enumeration by using appropriate techniques such as door-to-door survey, mail survey, telephone survey, serial survey, or information from local agricultural authorities or other reliable sources.

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

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less thant he 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 to unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 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 is MPC in air (submersion) was converted to a equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

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

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

3/4.11.1.3 LIQUID WASTE TREATMENT

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

3/4.11.1.4 LIQUID HOLDUP TANKS

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

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

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

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.

3/4.11.2.2 DOSE, NOBLE GASES

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

3/4.11.2.3 DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodologies provided in NUREG/CR-1004, " A Statistical Ananlysis of Selected Parameters for Predicting Food Chain Transport and Internal Dose of Radionuclides", October, 1979 and Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determineing the actual doses based upon the historical average atmospheric conditions. The release rate specifications for I-131, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent

3/4.11.2.3 DOSE - IODINE-131, TRITIUM, AND RADIONUCLIDES (Continued)

on the existing radionuclide pathways to man, in the unrestricted area. The pathways which 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 themilk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

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

3/4.11.2.6 GAS 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".

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

BASES

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a MEMBER OF THE PUBLIC for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

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

The LLDs required by table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement. 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.

3/4.12.2 LAND USE CENSUS

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

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

6.1.2 The Chief, Radiological Services, shall be responsible for implementing the radiological environmental program and dose calculations and projections as described in the Offsite Dose Calculation Manual (ODCM). These responsibilities include performance of surveillance requirements listed in Table 6.1-1.

6.1.3 The Shift Supervisor (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Site Director, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2-1.

6.2.1.2 The offsite organization for the radiological enviromental monitoring program and dose calculations shall be as shown in Figure 6.2-3.

UNIT STAFF

6.2.2 The Unit organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room.

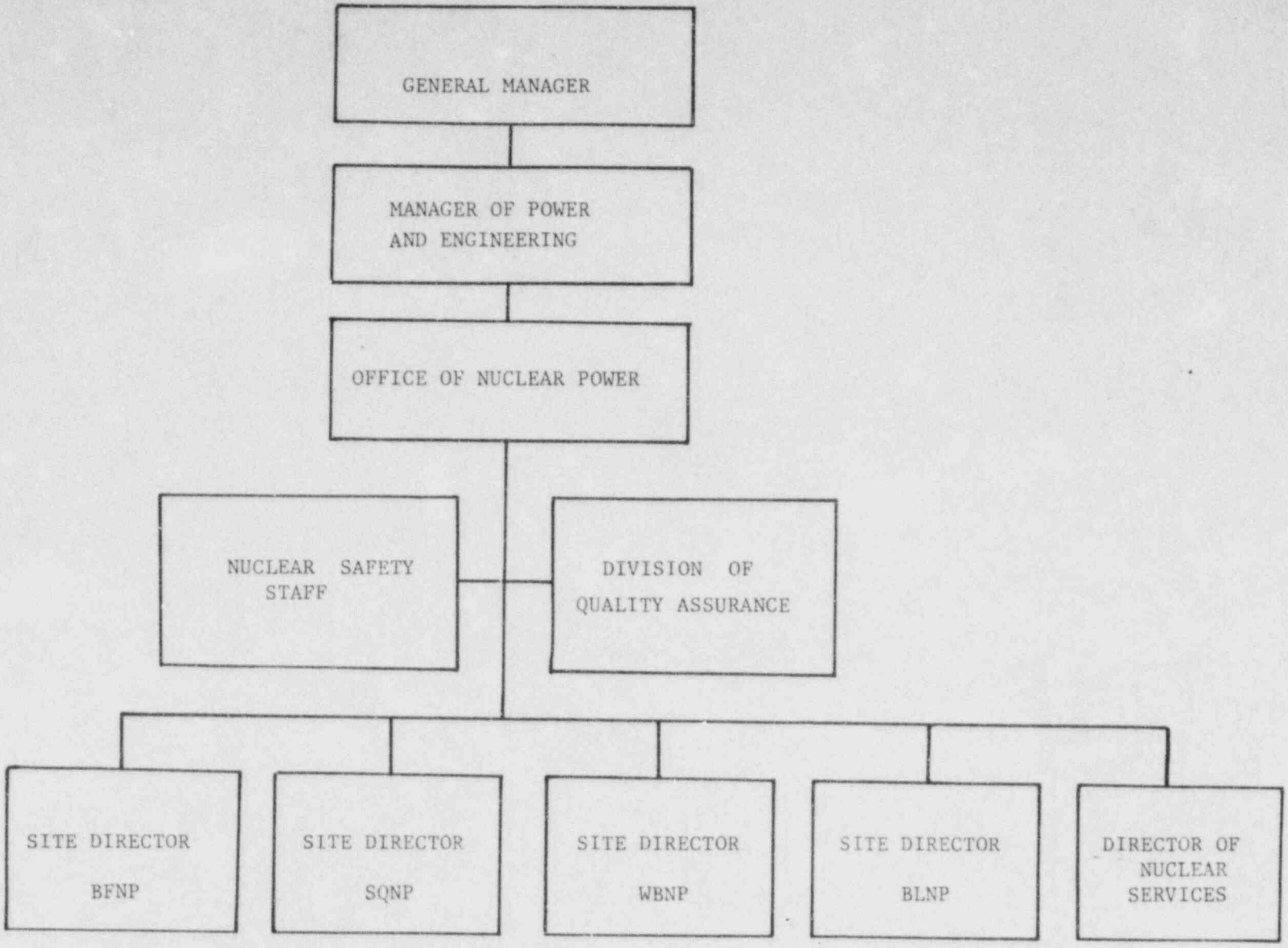
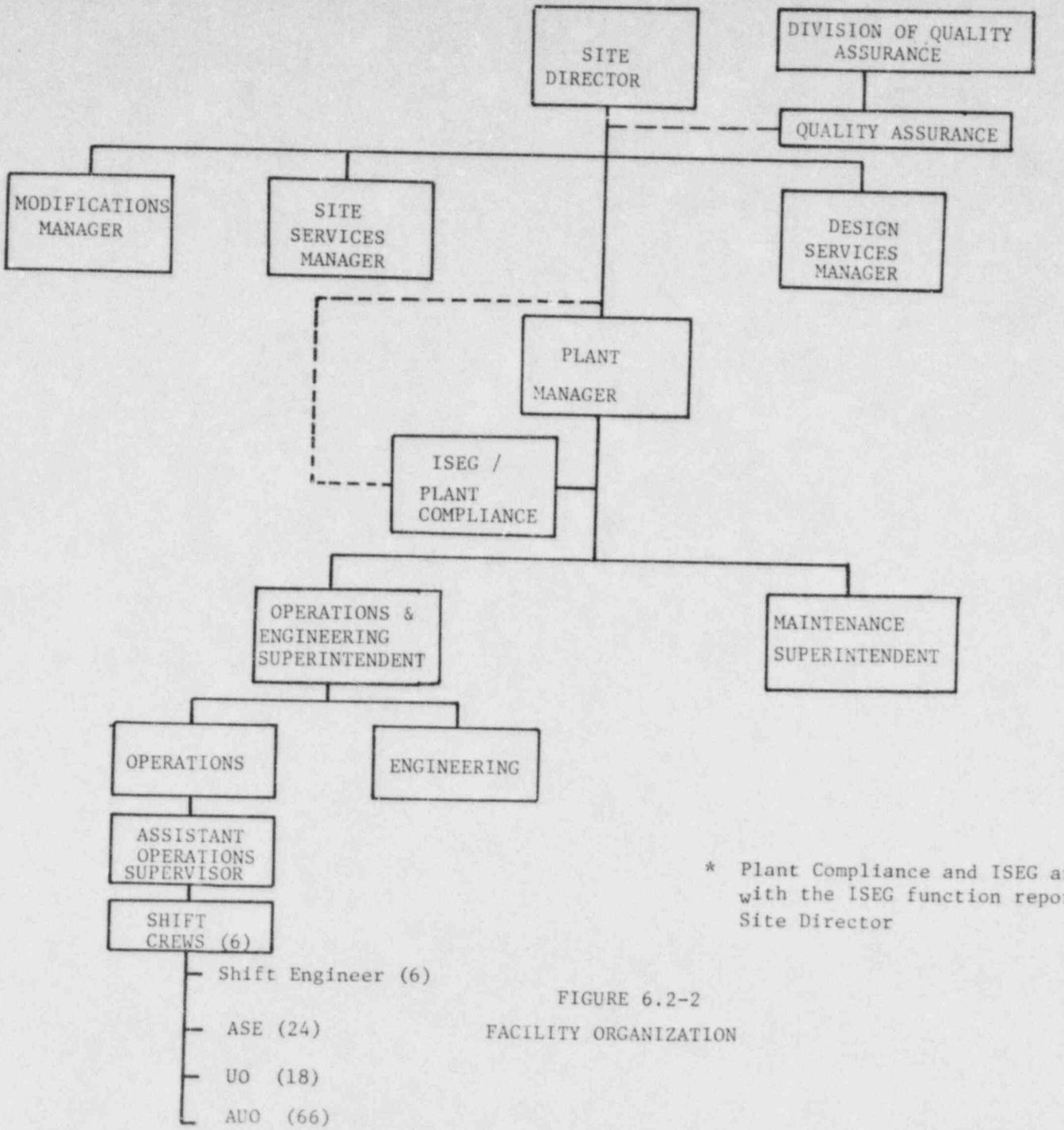


FIGURE 6.2-1

OFFSITE ORGANIZATION FOR FACILITY MANAGEMENT AND TECHNICAL SUPPORT



* Plant Compliance and ISEG are the same group with the ISEG function reporting to the Site Director

FIGURE 6.2-2

FACILITY ORGANIZATION

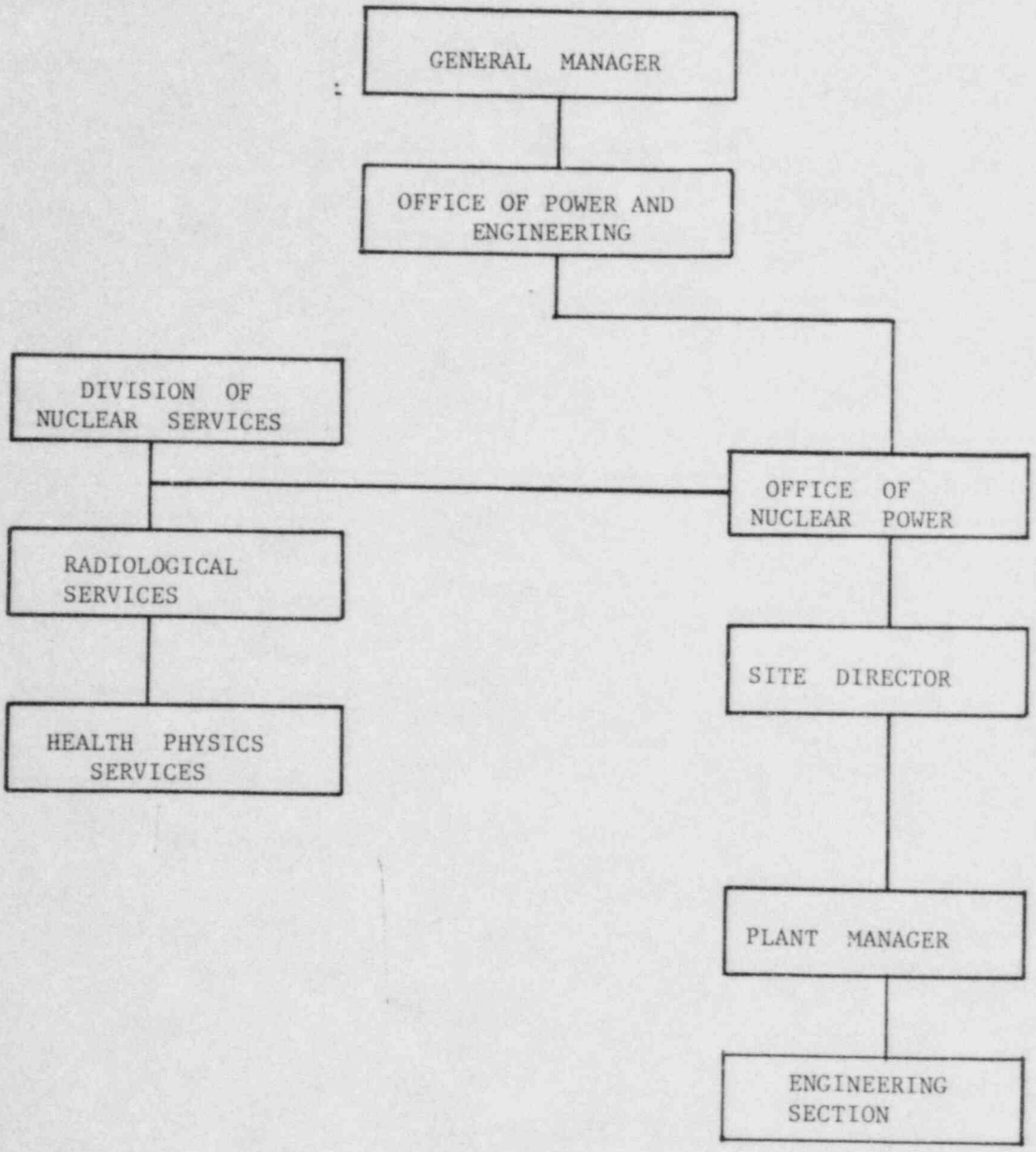


FIGURE 6.2-3

ORGANIZATION FOR MONITORING PROGRAM AND DOSE CALCULATIONS

TABLE 6.1-1

SURVEILLANCE REQUIREMENTS PERFORMED BY

HEALTH PHYSICS SERVICES

4.11.1.2
4.11.1.3.1
4.11.2.1.1 (partial)
4.11.2.1.2 (partial)
4.11.2.2
4.11.2.3
4.11.2.4
4.11.4
4.12.1
4.12.2
4.12.3

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensing Event Reports and other sources which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five full time engineers located onsite.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to the Site Director.

6.2.4 SHIFT TECHNICAL ADVISOR (STA)

6.2.4.1 The STA shall serve in an advisory capacity to the shift supervisor on matters pertaining to the engineering aspects of assuring safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of March 28, 1980 NRC letter to all licensees, except for the Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Operations and Engineering Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ISEG.

6.5 REVIEW AND AUDIT

6.5.0 The Manager of the Office of Nuclear Power is responsible for the safe operation of all TVA nuclear power plants. The functional organization for review and audit cognizance is shown on Figure 6.2-1.

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the Plant Manager matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Plant Manager
Member:	Operations Supervisor or Assistant Operations Supvr.
Member:	Engineering Group Supervisor or Engr. Section Supvr.
Member:	Maintenance Supervisor
Member:	Operations/Engineering Superintendent or Maintenance Superintendent
Member:	Health Physicist
Member:	Supervisor Quality Assurance

ALTERNATES

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

MEETING FREQUENCY

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

QUORUM

6.5.1.5 The minimum quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8.1 and changes thereto, 2) all programs required by Specification 6.8.5, and changes thereto, 3) any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

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

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

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

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Site Director and to the Chief of the Nuclear Safety Staff.

f. Review all written reports requiring 24 hour notification to the Commission.

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

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Chief, Nuclear Safety Staff.

i. Review of the Plant Physical Security Plan and implementing procedures and shall submit recommended changes to the Chief, Nuclear Safety Staff.

j. Review of the Site Radiological Emergency Plan and implementing procedures and shall submit recommended changes to the Chief, Nuclear Safety Staff.

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Director, Nuclear Power Division and to the Chief, Nuclear Safety Staff.

l. Review of changes to the radwaste treatment systems.

m. Review of meeting minutes of the Radiological Assessment Review Committee (RARC).

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend in writing to the Plant Manager approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

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

c. Provide written notification within 24 hours to the Site Director and the Chief, Nuclear Safety Staff of disagreement between the PORC and the Plant Manager ; however, the Plant Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Plant Manager and to the Chief, Nuclear Safety Staff.

6.5.2 NUCLEAR SAFETY Staff (NSS)

FUNCTION

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

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

RESPONSIBILITY

6.5.2.2 The NSS shall be responsible for the independent nuclear safety review program and cognizance of audits for all TVA nuclear plants.

QUALIFICATIONS

6.5.2.3 All NSS and other TVA personnel who have been designated as reviewers shall have an academic degree in engineering or a physical science field, or the equivalent; and in addition, shall have a minimum of five years of technical experience in one or more areas given in 6.5.2.1. The Chief, NSS shall meet the same qualifications as reviewers except that the minimum experience shall be six years.

CONSULTANTS

6.5.2.4 Consultants to the NSS shall be utilized to review or to provide expert advice as determined by the Chief, NSS.

MINIMUM REVIEW

6.5.2.5 A minimum of three reviewers shall review each of the subjects encompassed by sections 6.5.2.6 and 6.5.2.7.

REVIEW

6.5.2.6 The NSS shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

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

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All written reports requiring notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the PORC and the RARC.

AUDITS

6.5.2.7 Audits of unit activities shall be performed under the cognizance of the NSS. These audits shall encompass:

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

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

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

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

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

f. The Plant Physical Security Plan, the Safeguards Contingency Plan, and implementing procedures at least once per 12 months.

g. Any other area of unit operation considered appropriate by the NSS or the Manager of Nuclear Power.

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

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

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

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

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

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

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, December 1977 or Regulatory Guide 1.21 Rev.1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975, at least once per 12 months.

AUTHORITY

6.5.2.8 The NSS shall report to and advise the Manager of Nuclear Power areas of responsibility specified in Sections 6.5.2.6 and 6.5.2.7.

REPORTS

6.5.2.9 Reports of NSS activities shall be prepared, approved and distributed as indicated below:

a. Results of reviews and of cognizance of audits, including those encompassed by Sections 6.5.2.6 and 6.5.2.7 above, shall be approved by the Chief, NSS and forwarded to the Manager of Nuclear Power at least quarterly.

b. Audit reports encompassed by Section 6.5.2.7 above, shall be forwarded to the Manager of Nuclear Power and to the management positions responsible for the areas audited within 30 days after completion of the audit.

6.5.3 RADIOLOGICAL ASSESSMENT REVIEW COMMITTEE (RARC)

Function

6.5.3.1 The SQN RARC shall function to advise the Chief, Health Physics Services, and the Plant Manager on all matters related to radiological assessments involving dose calculations and projections and environmental monitoring.

Composition

6.5.3.2 The RARC shall be composed of the:

Chairman: Assessment Unit Supervisor, Health Physics Services
Member: Health Physicist, Gaseous, Health Physics Services
Member: Health Physicist, Liquid, Health Physics Services
Member: Meteorologist, Air Quality Branch
Member: Chemical Engineer, Engineering Section, SQN
Member: Office of Nuclear Power Representative

ALTERNATES

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

MEETING FREQUENCY

6.5.3.4 The SQN RARC shall meet at least once per six months or as requested by the SQN RARC Chairman or his/her designated alternate or a plant representative.

QUORUM

6.5.3.5 The minimum quorum of the SQN RARC necessary for the performance of the SQN RARC responsibility and authority provisions of these technical specifications shall consist of the Chairman or his designated alternate and 3 members (including alternates) as long as one is a plant representative.

RESPONSIBILITIES

6.5.3.6 The RARC shall be responsible for:

a. Review of changes to the OFFSITE DOSE CALCULATION MANUAL.

b. Review of all procedures required by Specification 6.8.4 and changes thereto.

c. Review for information purposes results of any audits, review or evaluation of the Quality Assurance Program for effluent and environmental monitoring and radiological assessments involving dose evaluations and projections.

d. Review of proposed changes to the technical specifications related to plant effluent radiological assessments involving dose calculations and projections and environmental radiological monitoring.

Authority

6.5.3.7 The RARC shall:

a. Recommend in writing to the Chief, Health Physics Services and to PORC approval or disapproval of items considered under 6.5.3.6 above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.3.6 constitutes an unreviewed safety question.

c. Provide timely written notification to the Site Director, and the Nuclear Safety Staff of unresolvable items associated with a and b above between the RARC and the Chief, Health Physics Services; however, the Chief, Health Physics Services, in coordination with the Plant Manager, shall have the responsibility for resolution of any such disagreement pursuant to 6.1.2 above.

Records

6.5.3.8 The SQN RARC shall maintain written minutes of each RARC meeting that at a minimum, document the results of all RARC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Site Director, PORC, and the Chief, Nuclear Safety Staff.

6.6 REPORTABLE EVENTS

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Sections 50.72 and 50.73 to 10 CFR Part 50 and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the report shall be submitted to the Chief, NSS and the Site Director.

6.7 SAFETY LIMIT VIOLATION

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

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

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Site Director and the Chief, NSS shall be notified within 24 hours.

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

d. The Safety Limit Violation Report shall be submitted to the Commission, the Chief, NSS and the Site Director within 14 days of the violation.

6.8 PROCEDURES AND PROGRAMS

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Plant Physical Security Plan implementation.

e. Site Radiological Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. Quality Assurance Program for effluent monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977 or Regulatory Guide 1.21 Rev.1, 1974 and Regulatory Guide 4.1, Rev. 1, 1975.

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

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

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

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

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

6.8.4 Written procedures shall be established, implemented and maintained by the Health Physics Services covering the activities below:

a. OFFSITE DOSE CALCULATIONAL MANUAL implementation.

b. Quality Assurance Program for environmental monitoring, using the guidance contained in Regulatory Guide 4.15, December 1977.

c. Surveillance requirements and environmental monitoring requirements shown in Table 6.1-1.

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

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the safety injection system, residual heat removal system, chemical and volume control system, containment spray system, and RCS sampling system. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

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

b. In-Plant Radiation Monitoring

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

(i) Training of personnel,

(ii) Procedures for monitoring, and

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

c. Secondary Water Chemistry

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

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

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

(iii) Identification of process sampling points,

ANNUAL REPORTS[1/]

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

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

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT[3/]

6.9.1.6 The radiological environmental operating report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an anlysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparision with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. and a listing of the new locations for dose calculations and/or environmental monitoring identified by the land use census. If harmfull effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problems and a planned course of action to alleviate the problem.

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

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

3/ 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 matrial from each unit.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

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

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT[1/]

6.9.1.8 The semiannual radioactive effluent release report covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

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

The annual radioactive effluent release report (Radiological Impact) to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same 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. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

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

The annual radioactive effluent release report to be submitted after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed members of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The semiannual radioactive effluent release reports shall include the following information for each type ;of solid waste identified in Regulatory 1.21, Rev.1, Table 3, Part A, which is shipped offsite during the report period:

a. Total volume of containers,

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

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

d. Type of quantity (e.g., LSA, Type A, Type B, etc.)

The semiannual radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis, and shall include any changes to the PROCESS CONTROL PROGRAM (PCP) and the Offsite Dose Calculation Manual (ODCM) made during the reporting period. It shall include the type of solidification agent used, if applicable.

MONTHLY REACTOR OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or Safety Valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the PORC.

6.10.1 (continued)

c. All REPORTABLE EVENTS submitted to the Commission.

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

e. Records of changes made to the procedures required by Specification 6.8.1 and 6.8.4.

f. Records of radioactive shipments.

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

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

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

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

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

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

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

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

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

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

i. Records of Quality Assurance activities required by the Operational Quality Assurance Manual.

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

k. Records of meetings of the PORC, RARC, and the NSS reports of the results of reviews and of cognizance audits,

l. Records of analyses required by the radiological environmental monitoring program.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.11 The W(z) function for normal operation shall be provided to the Director, Nuclear Reactor Regulation, 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. In the event that these values would be submitted at some other time during core life, it will be submitted 60 days prior to the date the values would become effective unless otherwise exempted by the Commission.

Any information needed to suport W(z) will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

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

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

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 Licensee initiated changes to the PCP:

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

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

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

   c. documentation of the fact that the change has been reviewed and found acceptable by the PORC.

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the next annual radioactive release report pursuant to specification 6.9.1.9. This submittal shall contain:

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

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

   c. documentation of the fact that the change has been reviewed and found acceptable by the RARC.

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

6.10.2 (Continued)

m. Records of secondary water sampling and water quality.

n. Records of the service life monitoring of all hydraulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b including the maintenance performed to renew the service life.

o. Records for environmental qualification which are covered under the provisions of paragraph 2.C.(12)(b) of license No. DPR-79.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

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

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

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

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Special (Radiation) Work Permit.

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

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

ENCLOSURE 2

TVA-SQN-TS-60

JUSTIFICATION FOR PROPOSED CHANGE TO TECHNICAL SPECIFICATIONS

SEQUOYAH NUCLEAR PLANT
UNITS 1 AND 2

ENCLOSURE

SEQUOYAH NUCLEAR

JUSTIFICATION FOR PROPOSED CHANGES
TO THE TECHNICAL SPECIFICATIONS

Item A. Radiological Effluent Technical Specification Changes

Description of Changes

Changes are made to the definitions, Limiting Conditions for Operation, action statements, reporting requirements, surveillance requirements, and administrative controls for the Radiological Effluent Technical Specifications (RETS).

Justification for Proposed Changes

Changes are being requested to reduce or eliminate unnecessary testing and sampling requirements while other changes are to modify overly restrictive action statements. The administrative controls are being changed to reflect TVA organizational changes and to eliminate unnecessary reporting requirements.

The majority of the changes to the RETS is based on information provided in NRC NUREG-0472, revision 3, issued March 30, 1982. Other changes are based on comments provided to NRC by the RETS subcommittee of the Atomic Industrial Forum under a letter from D. Harward dated June 8, 1982.

Changes to the secondary system sampling requirements were made based on changes approved by NRC to the draft Watts Bar Nuclear Plant (WBN) technical specifications. The Turbine Building sump and steam generator blowdown do not discharge directly to unrestricted areas, and their discharges are monitored continuously by radiation monitors. Maintaining the lower limit of detectibility for these monitors less than or equal to 10^{-6} μCi/ml and allowing for background will ensure that small amounts of primary to secondary leakage can be detected and the sampling programs initiated.

The changes to the containment venting sampling requirements are justified for several reasons. The containment vent path is from the lower containment to the containment annulus, then the containment annulus fans throgh the Auxiliary Building exhaust. The release pathway is monitored by the lower compartment Auxiliary Building exhaust radiation monitors. In addition, the Auxiliary Building exhaust and containment monitors provide isolation capability in the event of high radiation. The containment setpoint is typically set at 20 percent of the technical specification allowable value. The technical specification setpoint is based on compliance with 10 CFR Part 20 dose limits assuming continuous purge flow at 28,000 cfm for both

units simultaneously. Venting is performed several times each day during steady-state operation to ensure compliance with the normal containment internal pressure technical specifications. The typical vent volume is approximately 10,000 ft^3. The radioactivity concentrations and isotopic composition are essentially stable during this period. Daily sampling for isotopic analysis is sufficient to document radioactive release calculations during venting because of the stable reactor conditions and small release volumes.

Item B. Nuclear Safety Staff (NSS)

Description

This administrative change assigns responsibility for the independent review function to NSS rather than to NSRB. This change in organizational structure from a committee structure to an organizational unit structure results in changes to several sections of the technical specifications.

Justification for Proposed Changes

The reason for this change is to improve the depth and scope of the independent review function and to make it more administratively effective and consistent with administrative changes recently approved by POWER management.

ANSI N18.7-1976/ANS 3.2, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," as endorsed by NRC Regulatory Guide 1.33, Revision 2, approves two organizational structures for accomplishing the independent review function. This change to the technical specifications changes the organization structure of the independent review function from one approved structure to the other approved structure.

Item C. Office of Power Reorganization

Description of Change

This administrative change is being made to change the title of plant superintendent to plant manager.

The plant manager will report directly to the Site Director. The title of the Assistant Director of Nuclear Power (Operations) is being changed to Site Director.

The NSS (see Part B) will now be reporting directly to the Manager of the Office of Nuclear Power, rather than the Manager of Power.

Justification for Proposed Changes

Tennessee Valley Authority's (TVA) nuclear power program has gone through a major reorganization which gives added responsibility and resources to the individual nuclear power plants management.

The new organization framework has created a Manager of Power and Engineering in charge of the power program for TVA and a Manager of the Office of Nuclear Power in charge of the nuclear power program. The position of Site Director has been created at each nuclear power plant. The Site Director, who will report directly to the Office of Nuclear Power, will have overall responsibility for activities at Sequoyah Nuclear Plant (SQN).

The SQN Site Director has responsibility over five onsite organizations. They are:

1. Operations--This is supervised by a plant manager and encompasses most of the functions of the old plant organization. The plant manager supervises functions such as operations, maintenance, engineering, and compliance.

2. Modifications--This organization is responsible for carrying out major plant modifications.

3. Site Services--This group is responsible for management services, emergency planning coordination, regulatory engineering, and health physics support.

4. Design Services--This organization will provide long-term design engineering services.

5. Quality Assurance.

The functions of the existing Division of Nuclear Power Central Office Staff will be divided to retain a strong central office capability for standardization, technical support, and policy guidance to the nuclear plants, while transferring the direct support to the plants under the Site Director.

The Central Office engineering organization maintains responsibility for the integrity of engineering and design of SQN. It has responsibility for establishing and maintaining engineering and design standards, criteria and procedures, and provides centralized engineering support services to the plant that are not practical to be provided at the plant staff.

These changes provide the resources that are required for safe and efficient operation of SQN under a single line of authority and responsibility and provide more direct control by management at the site.

Item D. Independent Safety Engineering Group (ISEG)

Description of Changes

These changes are being requested to reflect the current status of the ISEG within the facility organization. Refer to revised figure 6.2-2. The ISEG will now report to the Site Director as a result of a recent TVA reorganization. Also a change is being requested to delete the term "dedicated" from 6.2.3.2.

The new organization framework has created a Manager of Power and Engineering in charge of the power program for TVA and a Manager of the Office of Nuclear Power in charge of the nuclear power program. The position of Site Director has been created at each nuclear power plant. The Site Director, who will report directly to the Office of Nuclear Power, will have overall responsibility for activities at Sequoyah Nuclear Plant (SQN).

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1. Operations--This is supervised by a plant manager and encompasses most of the functions of the old plant organization. The plant manager supervises functions such as operations, maintenance, engineering, and compliance.

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Justification for Proposed Changes

As required by operating license conditions, 2.C.(22).A for unit 1 and 2.C.(16).b for unit 2 and in accordance with the technical specifications for Sequoyah, TVA has an Independent Safety Engineering Group (ISEG). The ISEG consists of five permanently assigned engineers onsite who perform the ISEG function for both units. The ISEG functions are performed by the Plant Compliance Staff. As stated in the August 11, 1980 letter from L. M. Mills to A. Schwencer, "the Plant Compliance Staff will perform the function of engineering assessment and evaluation," as well as "dissemination of plant operating experiences."

Previously, the ISEG reported to the Assistant Director for Maintenance and Engineering, a high level manager located offsite. However, as a result of the recent TVA reorganization, the responsibility for control of all activities affecting plant quality assurance, financial planning, design engineering, scheduling and planning, plant performance, plant operational support, and regulatory performance, has been transferred to the site. Consistent with the TVA reorganization and plant site organization, the ISEG now reports to the Site Director. The Site Director is a high-level corporate manager located onsite, in a technically oriented position, and responsible for all activities affecting the plant. Refer to Item C for a description of the responsibilities. However, the Plant Manager remains directly responsible for day-to-day operation of the plant and reports to the Site Director.

Under the present organization and as previously indicated in the August 11, 1980 letter, the plant compliance staff will continue to perform the dual functions for the ISEG and the Compliance Staff. This combination of roles serves to enhance the Compliance Staff's ability to perform the ISEG function. The Compliance Staff functions of coordination of the plants investigation of a response to all inspection/audit findings, investigation of potential reportable occurrences (PROS), investigation and preparation of License Event Reports (LERs), tracking of corrective actions as well as trending of PROs, LERs, and NRC violations provides a broad overview of all activities potentially affecting plant safety. Thus, these functions serve to enhance the ISEG function. All violations, findings, and deviations written against the plant are channeled through one group and that group, Compliance Staff/ISEG, through the investigation and trending program, can identify the need for, evaluate the effectiveness of, and recommend corrective actions to the Site Director. Therefore, we see the roles of Compliance/ISEG as complimenting and also believe that a departure from the total dedication to ISEG functions, specified in NUREG-0737, to be justified.

The significant hazards consideration determination for item A, B, C, and D are provided in Attachments 1, 2, 3, and 4 respectively.

In summary, for Items A, B, C, and D based on the previously discussed justifications and the significant hazards consideration determinations, we have concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impace of the station on the environment as described in the NRC Environmental Statement.

ATTACHMENT 1

SIGNIFICANT HAZARDS CONSIDERATIONS
TECHNICAL SPECIFICATION CHANGE NO. 111, PART A

1. Is the probability of an occurrence or the consequences of an accident previously evaluated in the safety analysis report significantly increased? No.

   These changes primarily provide clarification in existing Radiological Effluent Technical Specifications (RETS) and increase operating flexibility in RETS guidelines recently provided by the Nuclear Regulatory Commission.

2. Is the possibility for an accident of a new or different type than evaluated previously in the safety analysis report created? No.

   No new accidents are created.

3. Is the margin of safety significantly reduced? No.

   Changes are in accordance with NRC guidelines. See 1. above.

ATTACHMENT 2

SIGNIFICANT HAZARDS CONSIDERATIONS
TECHNICAL SPECIFICATION CHANGE NO. 111, PART B

1. **Is the probability of an occurrence or the consequences of an accident previously evaluated in the safety analysis report significantly increased?** No.

   Technical specification section 6.5.2.

   The proposed technical specification change (Part B) is administrative and complies with the requirements of ANSI N18.7-1976/ANS-3.2 as endorsed by NRC Regulatory Guide 1.33 Revision 2.

2. **Is the possibility for an accident of a new or different type than evaluated previously in the safety analysis report created?** No.

   Same as 1. above.

3. **Is the margin of safety significantly reduced?** No.

   Same as 1. above.

ATTACHMENT 3

SIGNIFICANT HAZARDS CONSIDERATIONS
TECHNICAL SPECIFICATION CHANGE NO. 111, PART C

1. Is the probability of an occurrence or the consequences of an accident previously evaluated in the safety analysis report significantly increased? No.

The plant organization shown in Figures 6.1-1, -2, and -3 have been changed to reflect the recent reorganization. This change does not directly impact safety. These changes are administrative only in nature.

2. Is the possibility for an accident of a new or different type than evaluated previously in the safety analysis report created? No.

Same as 1. above.

3. Is the margin of safety significantly reduced? No.

Same as 1. above.

ATTACHMENT 4

SIGNIFICANT HAZARDS CONSIDERATIONS
TECHNICAL SPECIFICATION CHANGE NO. 111, PART D

1. Is the probability of an occurrence or the consequences of an accident previously evaluated in the safety analysis report significantly increased? No.

The plant organization shown in Figures 6.1-1, -2, and -3 have been changed to reflect the recent reorganization. This change does not directly impact safety. These changes are administrative only in nature.

2. Is the possibility for an accident of a new or different type than evaluated previously in the safety analysis report created? No.

Same as 1. above.

3. Is the margin of safety significantly reduced? No.

Same as 1. above.