UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

RANCHO SECO NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
License No. DPR-54

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Sacramento Municipal Utility District (the licensee) dated June 30, 1987, as supplemented October 3 and December 23, 1987 and January 11, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-54 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 98, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This amendment to the Technical Specification shall become effective within 30 days of the issuance date or prior to reactor criticality following the 1986/87 outage, whichever is first. The implementation delay is provided to allow time for modification of affected procedures and promulgation of these changes to personnel.

FOR THE NUCLEAR REGULATORY COMMISSION

George W. Knighton

George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 17, 1988

March 17, 1988

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-54

DOCKET NO. 50-312

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iia	iia
iii	iii
iv	iv
vi	vi
vii	vii
--	viia
viii	viii
ix	ix
x	x
xii	xii
1-6	1-6
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3-60	3-60
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ATTACHMENT TO LICENSE AMENDMENT NO.

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FACILITY OPERATING LICENSE NO. DPR-54

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RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

LIST OF TABLES

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LIST OF FIGURES (Continued)

1.11 FIRE SUPRESSION SYSTEMS

1.11.1 The FIRE SUPRESSION WATER SYSTEM shall consist of water sources, pumps and distribution piping with associated sectionalizing control of isolation valves. Such valves include yard hydrant valves and the first valve ahead of the water flow alarm device on each sprinkler header, hose standpipe or spray system riser which protect nuclear safety components.

1.11.2 The FIRE SUPRESSION CARBON DIOXIDE SYSTEM shall consist of a CO_2 source and distribution piping with sectionalizing control valves which protect nuclear safety components.

1.12 STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of:

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

b. The testing of one system, subsystem, train or designated components during each subinterval.

1.13 PROCESS CONTROL PROGRAM

PROCESS CONTROL PROGRAM (PCP) - The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION/DEWATERING of radioactive wastes from liquid systems is assured.

1.14 SOLIDIFICATION

SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints.

1.16 RESTRICTED AREA

That portion of the site property, the access to which is controlled by security fencing, equipment and personnel.

1.17 SITE BOUNDARY

Site Boundaries are defined by Figures 5.1-1 through 5.1-4.

1.18 DOSE EQUIVALENT I-131

The DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose via the inhalation pathway as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in 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."

1.19 MEMBER(S) OF THE PUBLIC

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

1.20 DEWATERING

The process which removes the slurry water from ion exchange resin or filter media that has been transferred to a disposal container in a manner which provides reasonable assurance that State, Federal and disposal site free standing liquid requirements are met.

1.21 MAXIMUM EXPOSED (HYPOTHETICAL) INDIVIDUAL

The MAXIMUM EXPOSED INDIVIDUAL is characterized as "maximum" with regard to food consumption, occupancy, and other usage or exposure pathway parameters in the vicinity of Rancho Seco that would represent an individual with habits greater than usually expected for the average of the population in general.

1.22 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL

The RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL shall be a manual containing the description of the Rancho Seco radiological environmental monitoring program. The REMP MANUAL shall contain a description of the environmental samples to be collected, the sample locations, sampling frequencies, and sample analysis criteria.

1.23 LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM

The LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM is the system designed and installed to reduce the quantity of radioactive materials in liquid effluents by collecting liquid effluent and providing processing for the purpose of reducing the total radioactivity prior to its release to the environment.

1.24 VENTILATION EXHAUST TREATMENT SYSTEM

The VENTILATION EXHAUST TREATMENT SYSTEMS are systems 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 SYSTEMS components.

1.25 PURGE - PURGING

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

1.26 VENTING

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

1.27 $\bar{E}$

$\bar{E}$-BAR shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of average beta and gamma energies per disintegration (in MEV) for isotopes with half lives greater than 20 minutes, making up at least 95% of the total activity in the coolant (excluding iodines).

TABLE 1.9-1

FREQUENCY NOTATION

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

3.1.4 REACTOR COOLANT SYSTEM ACTIVITY

Specification

3.1.4.1 The total fission product activity of the reactor coolant due to nuclides with half lives longer than 20 minutes shall not exceed $43/\bar{E}$ microcuries per gm whenever the reactor is critical. $\bar{E}$ is the average (mean) beta and gamma energies per disintegration, in MeV, weighted in proportion to the measured activity of the radionuclides in reactor coolant samples.

Bases

The above specification is based on limiting the consequences of a postulated accident involving the double-ended rupture of a steam generator tube. The rupture of a steam generator tube enables reactor coolant and its associated activity to enter the secondary system where volatile isotopes could be discharged to the atmosphere through condenser air-ejectors and through steam safety valves (which may lift momentarily). Since the major portion of the activity entering the secondary system is due to noble gases, the bulk of the activity would be discharged to the atmosphere. The activity release continues until the operator stops the leakage by reducing the reactor coolant system pressure below the set point of the steam safety valves and isolates the faulty steam generator. The operator can identify a faulty steam generator by using the off-gas monitors on the condenser air ejector lines; thus he can isolate the faulty steam generator within 34 minutes after the tube break occurred. During that 34 minute period, a maximum of 2740 ft^3 of hot reactor coolant will have leaked into the secondary system; this is equivalent to a cold volume of 1980 ft^3.

The controlling dose for the steam generator tube rupture accident is the whole-body dose resulting from immersion in the cloud of released activity. To insure that the public is adequately protected, the specific activity of the reactor coolant will be limited to a value which will insure that the whole-body annual dose at the site boundary will not exceed 0.5 rem, the limit in 10 CFR Part 20 for whole body dose in an unrestricted area.

Although only volatile isotopes will be released from the secondary system, the following whole-body dose calculation conservatively assumes that all of the radioactivity which enters the secondary system with the reactor coolant is released to the atmosphere. Both the beta and gamma radiation from these isotopes contribute to the whole-body dose. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employs the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. It is further assumed that meteorological conditions during the course of the accident correspond to Pasquill Type F and 0.6 meter per second wind speed, resulting in a X/Q value of 8.51×10^{-4} sec/m^3.

The combined gamma and beta whole body dose from a semi-infinite cloud is given by:

Dose (Rem) = $0.246 \cdot \bar{E} \cdot A \cdot V \cdot X/Q \cdot \rho$

$$A_{max}\ (\mu c/gm) = \frac{(\text{Dose})_{max}}{0.246 \cdot \bar{E} \cdot V \cdot X/Q \cdot \rho} = \frac{0.5}{0.246 \times \bar{E} \times 77.6 \times 8.51 \times 10^{-4} \times 0.713}$$

$$A_{max}\ (\mu c/gm) = 43/\bar{E}$$

Where

A = Reactor coolant activity (μCi/gm = mCi/Kgm)

V = Volume of hot reactor coolant leaked into secondary system (2740 ft^3 = 77.6 m^3)

X/Q = Atmospheric dispersion coefficient at site boundary for a two hour period (8.51×10^{-4} sec/m^3)

$\bar{E}$ = Average beta and gamma energies per disintegration (MeV)

ρ = Density of hot reactor coolant (0.713 gm/cc)

Calculations required to determine $\bar{E}$ will consist of the following:

A. Quantitative measurement of the specific activity (in units of μc/gm) of radionuclides with half lives longer than 20 minutes, which make up at least 95 percent of the total activity in reactor coolant samples.

B. A determination of the average beta and gamma decay energies per disintegration for each nuclide, measured in (A) above, by utilizing known decay energies and decay schemes (e.g., Table of Isotopes, Sixth Edition, March 1968).

C. A calculation of $\bar{E}$ by the average beta and gamma energy for each radionuclide in proportion to its specific activity, as measured in (A) above.

3.15 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.15-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.17.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology contained in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

Applicability During releases via the retention basin effluent discharge.

Action a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.17.1 are met, immediately 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 3.15-1.

Bases

During normal operations, all radioactive contaminated water from primary system leaks and drains (except demineralized reactor coolant as noted below) is processed in a liquid radwaste system and recycled into the Reactor Coolant Makeup System or otherwise reused in the controlled areas of the plant. Secondary system water is normally released from the plant.

If the secondary system water contains radioactive material it is first processed through the 'A' or 'B' Regenerant Hold-Up Tanks (RHUTs) and then transferred to the North or South Retention Basin. The water in a Retention Basin is released off-site as a batch release. These releases are monitored by the Retention Basin Effluent Discharge. During periods of primary to secondary leakage, or when the sumps are contaminated, administrative controls require the liquid effluent in turbine building sumps shall be diverted to the 'A' and 'B' Regenerant Hold-Up Tanks.

Demineralized reactor coolant can be transferred from the Demineralized Reactor Coolant Storage Tank (DRCST) to the 'A' and 'B' Regenerant Holdup Tanks for sampling, processing, and eventual discharge offsite as required by operational constraints.

Under normal conditions, the once through steam generators have no blow down. If a blow down is required during periods of primary to secondary leakage, all water will be retained and processed in the radwaste system or diverted to the 'A' and 'B' Regenerant Hold-Up Tanks.

3.15 (Continued)

Bases (Continued)

Radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of radioactive liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology contained in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of specification 3.17.1. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 3.15-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Retention Basin Effluent Discharge Monitor	1	With the monitor inoperable, effluent releases may be resumed provided that prior to initiating a release from the retention basin: 1. At least two independent samples are analyzed in accordance with Specification 4.21.1. 2. 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. Exert best efforts to return the monitor to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.2.3 why the inoperable monitor was not restored in a timely manner.

Table 3.15-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Flow Measurement Devices		
a. Regenerant Hold-Up Tank Discharge Line Total Flow	1	With the flow measurement device inoperable, releases to the retention basins may continue provided the total flow is estimated once every 4 hours by a tank level device.
b. Waste Water Flow Rate and Totalizer	1	With the flow measurement device inoperable, effluent releases via this pathway may continue provided the total flow is estimated at least once per 4 hours during retention basin releases by a level device in the discharge stream.

3.16 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.16-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.18.1a are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology contained in the ODCM. Continuous samples of the gaseous effluent for radioiodines and radioactive particulate material shall be taken as indicated in Table 3.16-1.

Applicability At all times.

Action a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Specification 3.18.1a are met, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.16-1. Exert best efforts to return the instrument to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report submitted pursuant to Specification 6.9.2.3 why the inoperability was not corrected in a timely manner.

Bases

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases radioactive of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology contained in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.18.1a. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The Auxiliary Building Stack is the effluent release point for the Waste Gas System. The Auxiliary Building Stack Noble Gas Activity monitor will perform the necessary Waste Gas System release termination. The monitor alarms and terminates a Waste Gas Decay Tank release automatically if the activity exceeds the setpoint limits.

The condenser air ejector exhaust has an individual noble gas monitor. This system exhausts into the Auxiliary Building ventilation system. Therefore, the Auxiliary Building Stack is the effluent release point and will alarm upon release of environmentally significant radioactive gases.

Fuel Storage Building exhaust is directed to the Auxiliary Building Stack where the exhaust is filtered and monitored for any activity prior to release to the atmosphere.

Table 3.16-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent		
a. Noble Gas Activity Monitor providing alarm and automatic termination of release. *	1	With the monitor channel alarm/trip setpoint less conservative than the setpoint calculated as described in the ODCM, immediately suspend the release or declare the channel inoperable. With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Table 4.22-1 within 24 hours. **
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Table 4.22-1 within 24 hours. **

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* See Table 3.5.5-1 for additional actions required for this monitor as an accident monitor.

** Interruption of continuous sampling is allowed for periods not to exceed one hour.

Table 3.16-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent (continued)		
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measurement Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

Table 3.16-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack		
a. Noble Gas Activity Monitor providing alarm and automatic termination of Waste Gas Header release*	1	With the monitor alarm/trip setpoint less conservative than the setpoint calculated as described in the ODCM immediately suspend the release or declare the channel inoperable. With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours. With the monitor inoperable, the contents of the Waste Gas System 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

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* See Table 3.5.5-1 for additional action required for this monitor as an accident monitor.

Table 3.16-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack (continued)		
		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.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Table 4.22-1 within 24 hours. **
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Table 4.22-1 within 24 hours. **
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measuring Devices	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

** Interruption of continuous sampling is allowed for periods not to exceed one hour.

Table 3.16-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
3. Auxiliary Building Grade Level Vent		
a. Noble Gas Activity Monitor *	1	With the monitor channel alarm/trip setpoint less conservative than the setpoint calculated as described in the ODCM, immediately suspend the release or declare the channel inoperable. With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Table 4.22-1 within 24 hours. **

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* See Table 3.5.5-1 for additional actions required for this monitor as an accident monitor.

** Interruption of continuous sampling is allowed for periods not to exceed one hour.

Table 3.16-1 (continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
3. Auxiliary Building Grade Level Vent (continued)		
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.**
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measurement Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

** Interruption of continuous sampling is allowed for periods not to exceed one hour.

3.17 LIQUID EFFLUENTS

3.17.1 Concentration

The concentration of radioactive material released in liquid effluents at any time beyond the Site Boundary For Liquid Effluents (see Figure 5.1-4) 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} μCi/ml total activity.

Applicability At all times

Action

With the concentration of radioactive material released from the site exceeding Specification 3.17.1, immediately restore concentration within the specification limits and report the event in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.2.3.

Bases

This Specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to areas beyond the Site Boundary For Liquid Effluent (see Figure 5.1-4) 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 outside the site will result in exposures within the limits of 10 CFR Part 20.106 to MEMBER(S) OF THE PUBLIC. 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.

3.17.2 Dose

The dose or dose commitment to a MAXIMUM EXPOSED (HYPOTHETICAL) INDIVIDUAL from radioactive materials in liquid effluents released beyond the Site Boundary For Liquid Effluents (see Figure 5.1.3) shall be limited to:

a. Less than or equal to 1.5 mrem to the total body and to less than or equal to 5.0 mrem to any organ during any calendar quarter; and,

b. Less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ during any calendar year.

Applicability At all times

Action a. With the calculated dose or dose commitment from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report pursuant to Specification 6.9.5. This Report will identify the cause(s) for exceeding the limit(s) and define the corrective actions to be taken to reduce the releases of radioactive material in liquid effluents and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

Bases

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." The dose calculation methodology 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 an individual 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. There is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in finished drinking water that are in excess of the requirements of 40 CFR 141.

3.17.3 Liquid Holdup Tanks

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

a. "A" and "B" Regenerant Holdup Tanks

b. Borated Water Storage Tank

c. Demineralized Reactor Coolant Storage Tank

d. Miscellaneous Water Holdup Tank

e. Outside Temporary Tanks

Applicability At all times

Action

With the quantity of radioactive material in any of the listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, and initiate actions to reduce the tank contents to within the limit. Reduce the tank contents to within the limit within the next 72 hours and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report.

Bases

Restricting the quantity of radioactive material contained in the specified outdoor tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentration at the nearest potable water supply and the nearest surface water supply in an unrestricted area would be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2. The limit applies to each tank individually.

Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system or the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM.

3.17.4 Liquid Effluent Radwaste Treatment

The LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the system shall be used to reduce the quantity of radioactive materials in liquid effluents prior to their discharge when projected doses due to the liquid effluent beyond the Site Boundary For Liquid Effluents (see Figure 5.1-3) when averaged over 31 days, would exceed 0.25 mrem to the total body or 0.83 mrem to any organ.

Applicability At all times.

Action

a. With the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.5 a Special Report which includes the following information:

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

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

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

Bases

The OPERABILITY of the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used, when specified, provides reasonable assurance that the releases of radioactive materials in liquid effluents are maintained "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. The specified limits governing the use of appropriate portions of the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM are the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3.18 GASEOUS EFFLUENTS

3.18.1 Dose Rate

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at or beyond the Exclusion Area Boundary (see Figure 5.1-1) shall be limited to the following values:

a. The dose rate limit for noble gases shall be less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin; and,

b. The dose rate limit for Iodine-131, Iodine-133, tritium, and for all radioactive materials in particulate form with half lives greater than 8 days shall be less than or equal to 1500 mrem/yr to any organ.

Applicability At all times

Action

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the limit(s) given in Specification 3.18.1 and report the event in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.2.3.

Bases

This specification is provided to ensure that the dose rate from gaseous effluents due to immersion or inhalation at any time at the Exclusion Area Boundary (Figure 5.1-1) will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentration 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 an individual in an unrestricted area to annual average concentrations exceeding the dose rate equivalent, on which the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106 (b)(1)) were derived. For individuals who may at times be within the Exclusion Area 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 Exclusion Area Boundary to less than or equal to 500 mrem/yr to the total body or to less than or equal to 3000 mrem/yr to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a person of any age group via the inhalation pathway to less than or equal to 1500 mrem/yr.

3.18.2 Dose-Noble Gases

The air dose due to noble gases released in gaseous effluents to areas at or beyond the Site Boundary For Gaseous Effluents (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter, to less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation; and,

b. During any calendar year, to less than or equal to 10 mrad for gamma radiation and to less than or equal to 20 mrad for beta radiation.

Applicability At all times

Action

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report pursuant to Specification 6.9.5. This Report will identify the cause(s) for exceeding the limit(s) and define the corrective action(s) taken to reduce the release of radioactive noble gases in gaseous effluents, and the corrective action(s) to be taken to assure that subsequent releases will be in compliance with the abovelimits.

Bases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive 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 an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents 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 that the air doses at the Site Boundary for Gaseous Effluents (Figure 5.1-3) are based upon the historical average atmospheric conditions.

3.18.3 Dose-Iodine-131, Iodine-133, Tritium and Radioactive Materials in Particulate Form.

The dose or dose commitment to a MAXIMUM EXPOSED (HYPOTHETICAL) INDIVIDUAL from Iodine-131, Iodine-133, tritium, and radioactive materials in particulate form with half-lives greater than eight days in gaseous effluents released to areas at or beyond the Site Boundary for Gaseous Effluents (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter, to less than or equal to 7.5 mrem to any organ; and,

b. During any calendar year, to less than or equal to 15 mrem to any organ.

Applicability At all times

Action

With the calculated dose or dose commitment from the release of Iodine-131, Iodine-133, tritium, and radioactive materials in particulate form with half-lives greater than eight days in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report pursuant to Specification 6.9.5. This Report will identify 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 release will be in compliance with the above annual limits.

Bases

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 Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. For individuals 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 dispersion factor above that for the restricted area boundary.

3.18.3 (continued)

Bases (continued)

The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977. These equations also provide for estimating doses based upon the historical average atmospheric conditions. For individuals 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 dispersion factor above that for the restricted area boundary.

The release rate specifications for radioiodines and radioactive materials in particulate form are dependent on the existing radionuclide pathways to man in areas at or beyond the Site Boundary for Gaseous Effluents (Figure 5.1-3). The pathways which were examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

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3.18.4 Gaseous Radwaste Treatment

The Waste Gas System and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of these systems shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected air doses due to gaseous effluent releases (see Figure 5.1-3), 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 from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.3 mrem to any organ.

Applicability

When Gaseous Radwaste Treatment System and/or Ventilation Exhaust Treatment System are not being used.

Action

a. With gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days a Special Report pursuant to Specification 6.9.5 which includes the following information:

   1. Explanation of why gaseous radwaste was being discharged without treatment, and identification of the equipment or subsystems not OPERABLE and the reason for inoperability.

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

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

Bases

The OPERABILITY of the Waste Gas System and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems is available for use whenever gaseous effluents require treatment prior to release to the environment. 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 are maintained "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems are the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3.18.5 Gas Storage Tanks

The quantity of radioactivity contained in each waste gas decay tank shall be limited to less than or equal to 135,000 curies of noble gases (considered as Xe-133).

Applicability At all times

Action

a. When the reactor coolant system activity reaches the limit of Specification 3.1.4, sample the on line waste gas decay tank daily to ensure that the 135,000 curie equivalent Xe-133 limit is not exceeded.

b. With the quantity of radioactive material in any waste gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.2.3.

Bases

Restricting the quantity of radioactivity contained in each waste gas decay tank provides assurance that in the event of an uncontrolled release of the tanks contents, the resulting total body exposure to an individual at the exclusion area boundary (See Figure 5.1-1) will not exceed 500 mrem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

Potential atmospheric releases from a waste gas decay tank are evaluated assuming design coolant activities (see page 14D-25 Vol. VI FSAR). Based on primary coolant activity as shown in Table 14D-7, the decay tank is assumed to hold the activity associated with the off-gas from one reactor coolant system degassing with no credit taken for decay.

Calculation of the limiting decay tank activity based on the coolant activity limit of Technical Specification 3.1.4 yields a maximum decay tank inventory of 98,414 Ci (Ref. FSAR Table 14D-23). In order for the decay tank inventory to reach the limiting condition for operation, coolant activity would have to exceed the Technical Specification 3.1.4 limit on coolant activity and this would require a reactor shutdown, thus preventing a further increase in gaseous activity.

Therefore, it is conservative to require that the online waste gas decay tank be sampled_daily upon reaching the reactor coolant system limiting activity value (43/E) to ensure the 135,000 curies equivalent Xe-133 is not exceeded. Once the coolant is below the limiting activity, there is no requirement to sample waste gas decay tanks except for discharging.

3.22 RADIOLOGICAL ENVIRONMENTAL MONITORING

The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.22-1.

Applicability At all times

Action

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.22-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.2.2, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, or seasonal unavailability.)

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting level of Table 3.22-2 when averaged over any calendar quarter, in addition to complying with the requirements of specification 3.25a, prepare and submit to the Commission within 30 days after the level of radioactivity has been determined, a Special Report pursuant to Specification 6.9.5 which includes an evaluation of any release conditions, environmental factors or other aspects which caused the reporting limits to be exceeded. This report will define corrective actions to reduce emissions such that potential exposures will meet Specifications 3.17.2, 3.18.2, and 3.18.3. When more than one of the radionuclides in Table 3.22-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 3.22-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.17.2, 3.18.2, and 3.18.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetation samples unavailable from any of the sample locations required by Table 3.22-1, identify the cause of the unavailability of samples and the locations for obtaining replacement samples in the next Annual Radiological Environmental Operating Report. The locations from which samples were unavailable may then be deleted from Table 3.22-1 provided the locations from which the replacement samples were obtained are added to the Radiological Environmental Monitoring Program as replacement locations, if available.

3.22 (continued)

Bases

The Radiological Environmental Monitoring Program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and ODCM modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The specified monitoring program is in effect at this time. Program changes may be initiated based on operational experience, and changes in regional population or agricultural practices. The sample locations have been listed in the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL to retain flexibility for making changes as needed.

The detection capabilities required in Table 4.26-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirement of 40 CFR 141.

Table 3.22-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
A. Radioiodine and Particulates	8	Continuous operation of sampler with sample collection as required by dust loading but at least once per week.	Radioiodine canister. Analyze at least once weekly for I-131. Particulate sampler. Analyze for Gross Beta radioact ity at leas: 2 hours following filter change. Perform gamma isotopic analysis on each sample where gross beta activity is greater than 10 times the yearly mean of control samples for the same sample period. Perform gamma isotopic analysis on composite (by location) for particulate filters sample at least once per quarter.
2. DIRECT RADIATION	At least 40 locations with 2 dosimeters at each location.	At least once per quarter.	Gamma dose. At least once per quarter.

* Sample locations are shown in the REMP MANUAL.

Table 3.22-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	1	Composite sample collected monthly	Gamma isotopic and tritium analysis of each composite.
	3	Grab sample collected monthly.	Gamma isotopic and tritium analysis of each sample.
b. Runoff	1	Grab sample collected fortnightly.	Gamma isotopic and tritium analysis of each sample.
c. Ground	2	At least once per quarter.	Gamma isotopic, and tritium analysis of each sample.
d. Mud and Silt	2	At least once semi-annually. One pint sample of the top 3" of material 2 ft. from shoreline.	Gamma Isotopic analysis of each sample.

* Sample locations are shown in the REMP MANUAL.

Table 3.22-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Minimum Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	4	At least weekly when animals are on pasture; at least once per month at other times.	Gamma isotopic analysis and I-131 analysis of each sample.
b. Fish and Invertebrates	3	At least quarterly. One sample of each species as listed in the REMP MANUAL.	Gamma isotopic analysis on edible portion of each sample.
c. Food	4	At time of harvest. One sample of each of the several classes of food products as shown in the REMP MANUAL.	Gamma isotopic analysis on edible portion of each sample.

*Sample locations are identified in the REMP MANUAL.

Table 3.22-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m³)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	2×10^4(a)				
Mn-54	1×10^3		3×10^4		
Fe-59	4×10^2		1×10^4		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
Zn-65	3×10^2		2×10^4		
Zr-Nb-95	4×10^2 (b)				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	2×10^2 (b)			300 (b)	
Gross beta	40	2			

(a) For drinking water samples, this is 40 CFR Part 141 value.
(b) Total for parent and daughter.

3.23 LAND USE CENSUS

A land use census shall be conducted annually 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 vegetation in each of the 16 meteorological sectors within a distance of five miles.

The Land Use Census shall also include information relevant to the liquid effluent pathway and gaseous effluent pathway such that the OFFSITE DOSE CALCULATION MANUAL (ODCM) and the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUAL can be kept current with the existing environmental and societal uses surrounding Rancho Seco.

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 values currently being calculated in Specifications 4.21.2, and 4.22.3, identify the new locations in the next Annual Radiological Environmental Operating Report.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.22, add the new location(s) to the Radiological Environmental Monitoring Program within 30 days or submit a Special Report to the Commission pursuant to Specification 6.9.5 that identifies the cause(s) for exceeding these requirements and the proposed corrective actions for precluding recurrence. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s) (via the same exposure pathway) may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. Identify the new location(s) in the next Annual Radiological Environmental Operating Report and also include in the report a revised figure(s) and table for the REMP manual reflecting the new location(s).

*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.

3.23 (Continued)

Bases

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the Radiological Environmental Monitoring Program and the ODCM are made if required by the results of this census. 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 vegetation will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used: (1) that 20 percent of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage); and (2) a vegetation yield of 2 kg/square meter.

In addition, by gathering information on the liquid effluent pathway and the gaseous effluent pathway, the census provides assurance that proper radiological environmental monitoring and radioactive effluent controls are in place for the adequate protection of the health and safety of the general public.

3.25 FUEL CYCLE DOSE

The dose or dose commitment to any real MEMBER OF THE PUBLIC due to releases of radioactive material in gaseous and liquid effluents and to direct radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which is limited to less than or equal to 75 mrem) in a calendar year.

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 3.17.2.a, 3.17.2.b, 3.18.2.a, 3.18.2.b, 3.18.3.a, or 3.18.3.b, or exceeding the reporting levels of Table 3.22-2, calculations shall be made including direct radiation contributions (including outside storage tanks, etc.) to determine whether the above limits of Specification 3.25 have been exceeded.

b. If the above limits have been exceeded, prepare and submit to the Commission within 30 days, a Special Report pursuant to Specification 6.9.5 that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, in a calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations.

c. If the estimated dose(s) exceed the above limits, and if the release condition resulting in the violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provision of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

Bases

This specification is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the numerical guides for design objective doses of Appendix I or exceeds the reporting levels for the Radiological Environmental Monitoring Program. For the Rancho Seco site it is unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the plant remains within twice the numerical guides for design objectives of 10 CFR 50 Appendix I and if direct radiation (outside storage tanks, etc.) is kept small. The Special

3.25 (Continued)

Bases (Continued)

Report will describe a course of action which should result in the limitation of the dose to a MEMBER OF THE PUBLIC for a calendar year 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. If the dose to any MEMBER OF THE PUBLIC is evaluated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190 is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation which is part of the uranium fuel cycle.

3.26 INTERLABORATORY COMPARISON PROGRAM

The contractor performing the analysis of radiological environmental monitoring samples for radioactive materials shall participate in an Interlaboratory Comparison Program approved by the Commission.

Applicability At all times

Action

With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.2.2.

Bases

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR 50.

Table 4.1-1 (Continued)
INSTRUMENT SURVEILLANCE REQUIREMENTS

	Channel Description	Check	Test	Calibrate	Remarks
42.	Reactor Building drain accumulation tank level	NA	NA	R	
43.	Incore neutron detectors	M(1)	NA	NA	(1) Check functioning, including functioning of computer readout and/or recorder readout.
44.	a. Process radiation monitoring system	W	Q	R	
	b. Area radiation monitoring system	W	M	Q	
	c. Chlorine Detector	W(1)	M(1)	R(1)	(1) Only required if the total quantity of gaseous chlorine in the RESTRICTED AREA exceeds 100 pounds.
	d. Containment Area Monitors	W	NA	R	
45.	Emergency plant radiation Instruments	M(1)	NA	R	(1) Battery check
46.	Environmental air monitors	M(1)	NA	R	(1) Check functioning
47.	Strong motion accelerometer	Q(1)	NA	R	(1) Battery check
48.	Deleted				
49.	Pressurizer Water Level	M	NA	R	
50.	Auxiliary Feedwater Flow Rate	M	NA	R	
51.	Spent Fuel Pool Level	W(1)	NA	R	(1) Daily during refueling when moving fuel or control rods.
52.	EMOV Power Position Indicator (Primary Detector)	M	NA	R	
53.	EMOV Position Indicator (Backup Detector) T/C or Acoustic	M	NA	R	
54.	EMOV Block Valve Position Indicator	M	NA	R	
55.	Safety Valve Position Indicator (Primary Detector) T/C	M	NA	R	

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

Item		Check	Frequency
1.	Reactor coolant	a. Radio-chemical analysis(1)	M
		Ē determination (3)(4)(6)	Semiannually
		b. Gross activity(1) (3)	3/week
		c. Tritium radioactivity	M
		d. Chemistry (Cl and O_2)	3/week
		e. Boron concentration	2/week
		f. Fluoride	M
2.	Borated water storage tank water sample	Boron concentration(5)	M and after each makeup
3.	Core flooding tank water sample	Boron concentration(3)	M and after each makeup
4.	Spent fuel storage water sample	Boron concentration	M and after each makeup
5.	Secondary coolant	a. Gross activity(3)	Weekly
		b. Iodine analysis(2)(3)	Weekly
6.	Concentrated boric acid tank	Boron concentration(5)	2/week and after each makeup
7.	Spray additive tank	NaOH concentration (3) each makeup	Q and after each makeup
8.	Cooling Tower water	Gross activity (3)	M

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

Table Notation

(1)When radioactivity level is greater than 20 percent of the limits of Technical Specification 3.1.4, the sampling frequency shall be increased to a minimum of once each day.

(2)When gross activity increases by a factor of two above normal, an iodine analysis will be made and performed thereafter when the gross activity increases by ten percent.

(3)Not performed during cold shutdown.

(4)$\bar{E}$ determination will be started when a gross activity analysis indicates greater than 10μCi/gm. $\bar{E}$ will be redetermined each 10μCi/gm increase in gross activity. A radio chemical analysis for this purpose shall consist of a quantitative measurement of 95% of radionuclides in reactor coolant with half lives of >20 minutes.

(5)Not required during periods when systems are shutdown for maintenance.

(6)Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

4.19 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

The maximum setpoint shall be determined in accordance with methodology as described in the Offsite Dose Calculation Manual (ODCM) and shall be recorded on the release permits.

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the INSTRUMENT CHANNEL CHECK, SOURCE CHECK, INSTRUMENT CHANNEL CALIBRATION, AND CHANNEL TEST at the frequencies shown in Table 4.19-1.

Records shall be maintained in accordance with the Process Standards of all radioactive liquid effluent monitoring instrumentation alarm/trip setpoints. Maximum setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.17.1 are met.

Bases

The radioactive liquid effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential release of radioactive liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology contained in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.17.1. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 4.19-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
1. Gross Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Retention Basin Effluent Discharge Monitor	D(1)	P	R(2)	Q(3)
2. Flow Monitors				
a. Regenerant Hold-up Tank Discharge Line Total Flow	D(4)	NA	R	Q
b. Waste Water Flow Rate and Totalizer	D(4)	NA	R	Q

TABLE 4.19-1 (Continued)

TABLE NOTATION

(1) During releases via this pathway, a check shall be performed at least once per 24 hours.

(2) The instrument Channel Calibration for radioactivity measurement instrumentation shall be performed using one or more reference standards.

(3) The Channel Test shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

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

b. Circuit failure.

c. Instrument indicates a downscale failure.

d. Instrument controls do not set in operate mode.

(4) The Instrument Channel Check shall consist of verifying indication of flow during periods of release. The Instrument Channel Check shall be made at least once daily on any day on which batch releases are made.

4.20 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

The maximum setpoints shall be determined by procedures implementing the methodology described in the OFFSITE DOSE CALCULATION MANUAL (ODCM) and shall be recorded on release permits.

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the INSTRUMENT CHANNEL CHECK, SOURCE CHECK, INSTRUMENT CHANNEL CALIBRATION, AND CHANNEL TEST at the frequencies shown in Table 4.20-1.

Records shall be maintained in accordance with the Process Standards of all radioactive gaseous effluent monitoring instrumentation alarm/trip setpoints. Maximum setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.18.1 are met.

Bases

The radioactive gaseous effluent monitoring instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of radioactive gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology contained in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of Specification 3.18.1. The OPERABILITY and use of this instrumentation is consistent with the requirements and General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The flow rates in the Auxiliary Building Stack and Auxiliary Building Grade Level Vent are constant as they use single speed fans. The Reactor Building Purge Vent has a constant release rate. However, releases from the Reactor Building may be at three different flowrates, winter, summer or minipurge. Administrative controls assure that the correct flowrate is used.

The flow rates of the ventilation systems are periodically determined by surveillance procedures. The flow rate devices must be removed from the ventilation systems for the channel test, and in addition transported to the manufacturer for calibration. The frequencies have been set as shown in Table 4.20-1.

Table 4.20-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
1. Reactor Building Purge Vent				
a. Noble Gas Activity Monitor	D	M(4)	R(3)	Q(1)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device	D	NA	R	Q(6)
e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q
2. Auxiliary Building Stack				
a. Noble Gas Activity Monitor	D(5)	M	R(3)	Q(7)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device	D	NA	R	Q(6)
e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q

Table 4.20-1 (Continued)

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
3. Auxiliary Building Grade Level Vent				
a. Noble Gas Activity Monitor	D	M	$R^{(3)}$	$Q^{(2)}$
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device	D	NA	R	Q
e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q

Table 4.20-1 (Continued)

TABLE NOTATION

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The INSTRUMENT CHANNEL CALIBRATION shall be performed using one or more reference standards.

(4) A check shall be performed prior to each release.

(5) A check shall be performed prior to each release via a Waste Gas Decay Tank(s).

(6) To be performed when device is accessible and conditions do not pose a personnel safety hazard (i.e., potential main steam safety actuation).

(7) The CHANNEL TEST shall also demonstrate that the Waste Gas System automatically isolates and that control room annunciation occurs if any of the following conditions exist:

1. Instrument indictes measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

4.21 LIQUID EFFLUENTS

4.21.1 Concentration

Surveillance Requirements

The concentration of radioactive material at any time in liquid effluents released from the site shall be continuously monitored in accordance with Table 3.15-1.

The liquid effluent continuous monitor having provisions for automatic termination of liquid releases, as listed in Table 3.15-1, shall be used to limit the concentration of radioactive material released at any time from the site to areas beyond the site boundary to the limits given in Specification 3.17.1.

The radioactivity content of each batch of liquid effluent to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 4.21-1. The results of pre-release analyses shall be used with the calculational methods in the OFFSITE DOSE CALCULATION MANUAL (ODCM) to assure that the concentration at the point of release is limited to the limits of Specification 3.17.1.

Bases

This Specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to areas beyond the site boundary for liquid effluent 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 outside the site will result in exposures within the limits of 10 CFR Part 20.106 to MEMBER(S) OF THE PUBLIC. 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.

There are no continuous releases of radioactive material in liquid effluents from the plant. All radioactive liquid effluent releases from the plant are by batch method.

Table 4.21-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit Of Detection (LLD) (uCi/ml)(a)
A. Batch Waste Release Tanks(b,d)	Each Batch P	Each Batch P	Principal Gamma Emitting Nuclides (c)	2E-8
			I-131	6E-8
			Dissolved and Entrained Gases (Gamma Emitters)	1E-5
			H-3	1E-5

TABLE 4.21-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

a. (1) The lower limit of detection (LLD) for a radionuclide presented in this table is the smallest concentration, expressed in microcuries per milliliter, which is required to be detected, if present, in order to achieve compliance with the limits of Specification 3.17.1 (10CFR20, Appendix , Table II, Column 2).

(2) The LLD of a radioanalysis system is that value which will indicate the presence or absence of radioactivity in a sample when the probability of a false positive and of a false negative determination is stated. The probabilities of false positive and false negative are taken as equal at 0.05. The equation for LLD in microcuries per milliliter is given by the equation:

$$LLD = \frac{2.71 + 3.29(Br)^{0.5}}{3.70E4(YEVT)}$$

where 2.71 = factor to account for Poisson distribution at very low background count rates.

When background is estimated from a blank which has been counted for a specific period, the following applies:

B = estimated background (counts)

$$= b \ \frac{t_s}{t_b}$$

b = blank background (counts)
t_b = blank count time (seconds)
t_s = sample count time (seconds)

$$r = 1 + \frac{t_s}{t_b}$$

TABLE 4.21-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

3.70E4 = disintegrations/second/microcurie

Y = yield of radiochemical process, i.e., the product of all factors such as emission fraction, chemical yield, etc.

E = counting efficiency (count/disintegrations)

V = sample volume (milliliters)

$$T = \frac{[1-\exp(-\lambda t_s)]\exp(-\lambda t_c)}{\lambda}$$

where λ = decay constant (seconds^{-1})
t_c = time from midpoint of collection to start of counting

(3) When spectroscopy is used to analyze the sample, the following LLD equation is used:

$$LLD = \frac{2.71 + 4.65(B)^{0.5}}{3.70E4\ (YEVT)}$$

Where B is the counts in the Region Of Interest.

(4) The LLD is defined as an **a priori** (before the fact) limit and not as an **a posteriori** (after the fact) limit.

TABLE 4.21-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

b. Prior to discharge from the 'A' or 'B" RHUT, samples are collected and analyzed for accountability of activity in the Retention Basins. Prior to sampling the RHUTs, each batch will be isolated, and then thoroughly mixed, to assure representative sampling. A batch release is the discharge of liquid wastes of discrete volume from the north or south Retention Basin. Samples will also be collected from the Retention Basin and analyzed prior to discharge.

c. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-136, Cs-137, C3-141, Ce-144, and Ba-140. Other peaks which are measureable and identifiable, together with the listed nuclides, shall also be identified and reported. Nuclides which are not observed for the analysis shall be reported as "less than" the instrument's LLD, and shall not be reported as being present. The "less than" values shall not be used in the ODCM evaluations. However, if the nuclide is measured and identified at a value less than the Table 4.21-1 LLD value, it shall be reported and entered into the ODCM evaluations.

d. Miscellaneous Water Evaporator release is via the gaseous pathway.

4.21.2 Doses

Dose Calculations

Cumulative dose contributions and cumulative dose projections associated with the release of liquid RADIOACTIVE EFFLUENTS from the site (see Figure 5.1-3) shall be determined in accordance with the sampling and analyses specified in Tables 4.21-1 and 4.21-2 and the methodology described in the Offsite Dose Calculation Manual (ODCM) at the following frequencies:

a. Prior to the initiation of a release of liquid RADIOACTIVE EFFLUENT and, a dose calculation update shall be made; and,

b. Monthly, based on gamma-emitter and tritium analyses of liquid RADIOACTIVE EFFLUENT releases during the previous calendar month and the results of analyses performed on composite samples shall be added to the monthly dose calculation.

A dose tracking system and administrative dose limits shall be established and maintained. Operating parameters shall be adjusted in accordance with methodology described in the ODCM such that the dose values at any time, when projected to the end of the applicable time period, do not exceed the doses specified in Technical Specification 3.17.2.

Bases

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. Specification 4.21.2 provides the required operating flexibility and, at the same time, implements the guides set forth in Section IV.A of Appendix I which assures, by definition, that the releases of radioactive material in liquid effluents will be kept "as low as reasonably achievable." The dose calculations methodology in the ODCM implements 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 an individual 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 Dose to Man from Routine Releases of Reactor Effluent for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977, and Regulatory Guide 1.13, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

The results from composite samples during the period 1981 through 1984 indicates that Cs-137, Cs-134, Co-58 and Co-60 constitute 80 percent of the historical mix of gamma emitting radionuclides in plant liquid effluents. Another 13 percent consists of I-131. When the thyroid is separated as a limiting organ, 97.8 percent of the total body dose and 97.6 percent of the limiting organ dose are due to Cs-134 and Cs-137. Essentially 100 percent of the thyroid dose is due to I-131.

The activity analysis of Cs-134, Cs-137 and I-131 at the Lower Limits of Detection specified in Table 4.21-1 are based on an estimated annual plant radioactive effluent outflow of 20 million gallons per year with a minimum average dilution flowrate of 8,500 gallons per minute. These Lower Limits of Detection equate to an offsite dose commitment of approximately 50 percent of the guidelines specified in 10CFR50, Appendix I and provide an adequate basis for determining the presence or absence of dose due to other radionuclides in plant liquid effluents, when no other indications are revealed during sample analysis. The Lower Limits of Detection specified in Table 4.21-2 equate to an offsite dose commitment of approximately 10 percent of the 10CFR50, Appendix I guidelines.

The dose tracking system ensures that the dose limits prescribed in Technical Specification 3.17.2 will not be exceeded at the 95 percent confidence level. The methodology presented in the ODCM provides for adjustment of operational and analysis parameters to factor in variables such as annual radiological liquid effluent release volume, discharge canal flow rate, and current cumulative dose.

The dose tracking system provides for prompt updating of cumulative dose and contains feedback mechanisms to assure that the 10 CFR 50, Appendix I design objectives are not exceeded.

There is also reasonable assurance that the operation of the facility will not result in radionuclide concentrations in finished drinking water that are in excess of the requirement of 40CFR141.

Table 4.21-2

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis (c)	Lower Limit Of Detection (LLD) (uCi/ml)(a)
A. Batch Waste Release Tanks(b)	Each Batch P	Composite (d) M	Principal Gamma Emitting Nuclides (c)	4E-9
			H-3	1E-5
			Gross Alpha	1E-7
			Sr-89, Sr-90	3E-8

TABLE 4.21-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

a. (1) The lower limit of detection (LLD) for a radionuclide presented in this table is the smallest concentration, expressed in microcuries per milliliter, which is required to be detected, if present, in order to achieve compliance with the limits Of Specification 3.17.2 (10CFR50, Appendix I).

(2) The LLD of a radioanalysis system is that value which will indicate the presence or absence of radioactivity in a sample when the probability of a false positive and of a false negative determination is stated. The probabilities of false positive and false negative are taken as equal at 0.05. The equation for LLD in microcuries per milliliter is given by the equation:

$$LLD = \frac{2.71 + 3.29(Br)^{0.5}}{3.70E4(YEVT)}$$

where 2.71 = factor to account for Poisson distribution at very low background count rates.

When background is estimated from a blank which has been counted for a specific period, the following applies:

B = estimated background (counts)

$$= b \frac{t_s}{t_b}$$

b = blank background (counts)
t_b = blank count time (seconds)
t_s = sample count time (seconds)

$$r = 1 + \frac{t_s}{t_b}$$

TABLE 4.21-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

3.70E4 = disintegrations/second/microcurie

Y = yield of radiochemical process, i.e., the product of all factors such as emission fraction, chemical yield, etc.

E = counting efficiency (count/disintegrations)

V = sample volume (milliliters)

$$T = \frac{[1-\exp(-\lambda t_s)]\exp(-\lambda t_c)}{\lambda}$$

where λ = decay constant (seconds $^{-1}$)
t_c = time from midpoint of collection to start of counting

(3) When spectroscopy is used to analyze the sample, the following LLD equation is used:

$$LLD = \frac{2.71 + 4.65(B)^{0.5}}{3.70E4\ (YEVT)}$$

Where B is the counts in the Region Of Interest.

(4) The LLD is defined as **a priori** (before the fact) limit and not as an **a posteriori** (after the fact) limit.

TABLE 4.21-2 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

b. A batch release is the discharge of liquid wastes of discrete volume from the north or south retention basin. Prior to sampling the RHUTs, each batch will be isolated, and then thoroughly mixed, to assure representative sampling. Samples will also be collected from the Retention Basins and analysed prior to discharge.

c. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-136, Cs-137, Ce-141, Ce-144, and Ba-140. Other peaks which are measureable and identifiable, together with the listed nuclides, shall also be identified and reported. Nuclides which are not observed for the analysis shall be reported as "less than" the instrument's LLD, and shall not be reported as being present. The "less than" values shall not be used in the ODCM evaluations. However, if the nuclide is measured and identified at a value less than the Table 4.21-2 LLD value, it shall be reported and entered into the ODCM evaluations.

d. A composite sample is one in which the quantity of liquid samples 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.

4.21.3 Liquid Holdup Tanks*

Surveillance Requirements

The quantity of radioactive material contained in each tank listed in Specification 3.17.3 shall be determined to be within the specified limit by analyzing a representative sample of the tank's contents at least weekly when radioactive materials are being added to the tank.

Bases

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

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system or the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM.

4.21.4 Liquid Effluent Radwaste Treatment

Surveillance Requirements

Doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM) when LIQUID EFFLUENT RADWASTE TREATMENT SYSTEMS are not being fully utilized. The installed LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM shall be considered OPERABLE by meeting Specifications 3.17.1 and 3.17.2.

Bases

The OPERABILITY of the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the release of radioactive materials in liquid effluents will be kept "as low as is reasonable 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 the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

4.22 GASEOUS EFFLUENTS

4.22.1 Dose Rate

Surveillance Requirements

The noble gas effluent continuous monitors, as listed in Table 3.16-1, shall use monitor setpoints to limit the dose rate in unrestricted areas to the limits in Specification 3.18.1.

In the event a noble gas effluent exceeds the setpoint of its monitor, an assessment of compliance with Specification 3.18.1a shall be made in accordance with the methodology described in the ODCM.

The release rate of radioactive materials, other than noble gases, in gaseous effluents shall be determined by obtaining representative samples and performing analyses in accordance with the sampling and analysis program, specified in Table 4.22-1.

The dose rate due to Iodine-131, Iodine-133, tritium, and all radioactive material in particulate form with half-lives greater than 8 days released in gaseous effluents, shall be determined to be within the limits in Specification 3.18.1 by using the results of the sampling and analysis program specified in Table 4.22-1, and in accordance with the methodology described in the ODCM.

Bases

This specification is provided to ensure that the dose rate at any time at the Exclusion Area Boundary (Figure 5.1-1) from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)(1)). For individuals who may at times be within the Exclusion Area 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 Exclusion Area Boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3,000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to any person via the inhalation pathway to less than or equal to 1,500 mrem/year at the Exclusion Area Boundary.

Table 4.22-1

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (a) (uCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters (f)	1×10^{-4}
B. Reactor Building Purge Vent	P Each Purge Grab Sample(b,e,i)	P Each Purge (b,e,i)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
C. Auxiliary Building Stack	M(b,c,e) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
D. Auxiliary Building Grade Level Vent	M(b) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
E. All Release Types as listed in A,B,C,D above	Continuous	W(d) Charcoal Sample	I-131	1×10^{-12}
	Continuous	W(d) Particulate Sample	Principal Gamma Emitters (f) (I-131, Others)	1×10^{-11}
	Continuous	M Composite Particulate Sample	Gross Alpha(h)	1×10^{-11}
			Sr-89, Sr-90(g)	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1×10^{-4} as Xe^{-133}

Table 4.22-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

a. (1) The lower limit of detection (LLD) for a radionuclide presented in this table is the smallest concentration, expressed in microcuries per unit volume, which is required to be detected, if present, in order to achieve compliance with the limits of Specifications 3.18.1, 3.18.2 and 3.18.3.

(2) The LLD of a radioanalysis system is that value which will indicate the presence or absence of radioactivity in a sample when the probability of a false positive and of a false negative determination is stated. The probabilities of false positive and false negative are taken as equal at 0.05. The equation for LLD in microcuries per cubic centimeter is given by the equation:

$$LLD = \frac{2.71 + 3.29(Br)^{0.5}}{3.70E4(YEVT)}$$

where 2.71 = factor to account for Poisson distribution at very low background count rates.

When background is estimated from a blank which has been counted for a specific period, the following applies:

B = estimated background (counts)

$$= b \frac{t_s}{t_b}$$

b = blank background (counts)
t_b = blank count time (seconds)
t_s = sample count time (seconds)

$$r = 1 + \frac{t_s}{t_b}$$

Table 4.22-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

3.70E4 = disintegrations/second/microcurie

Y = yield of radiochemical process, i.e., the product of all factors such as emission fraction, chemical yield, etc.

E = counting efficiency (count/disintegrations)

V = sample volume (cubic centimeters)

where λ = decay constant (seconds $^{-1}$)
t_c = time from midpoint of collection to start of counting

(3) When spectroscopy is used to analyze the sample, the following LLD equation is used:

$$LLD = \frac{2.71 + 4.65(B)^{0.5}}{3.70E4\ (YEVT)}$$

Where B is the counts in the Region Of Interest.

(4) The LLD is defined as an **a priori** (before the fact) limit and not as an **a posteriori** (after the fact) limit.

b. Analysis shall also be performed when gross beta or gamma activity anaylsis of reactor coolant indicates greater than 10 µCi/ml. The analysis shall be repeated after each additional increase of 10 µCi/ml in the reactor coolant gross beta or gamma activity analysis.

c. Tritium grab samples shall be taken at least once per seven days from the ventilation exhaust from the auxiliary building stack during refueling and anytime fuel is in the spent fuel pool and the pool temperature exceeds 110°F. Below 110°F there is essentially no evaporation from this source.

Table 4.22-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

d. Samples shall be changed at least weekly and analyses shall be completed within 48 hours. Sampling and analysis shall also be performed when reactor coolant indicates 10μCi/ml gross beta gamma activity and every 10μCi/ml increases thereafter. When samples collected for less than 24 hours are analyzed, the corresponding LLDs maybe increased by a factor of 10.

e. Tritium grab samples shall be taken at least daily during refueling activities.

f. Principal gamma emitters for which the LLD applies are: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, Xe-135m, and Xe-138 for gaseous samples and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, (or Tc99m), Cs-134, Cs-137, Ce-141, and Ce-144 for particulate samples. This list does not mean only these nuclides will be detected and reported. Other peaks that are measurable and identifiable shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.2.3. Nuclides which are below the LLD for the analysis shall be reported as "less than" the nuclide's LLD and shall not be reported as being present at the LLD level for that nuclide. However, if the nuclide is measured and identified at a value less than its predetermined LLD value, it shall be reported and entered into the ODCM evaluations.

g. Gross beta analysis performed on a monthly basis for each environmental release particulate sample. If any one of these samples indicates greater than 1.0 E-11 μCi/cc gross beta activity then a Sr-89, Sr-90 analysis will be performed on those samples exceeding this value.

h. Gross alpha performed on a monthly basis for each environmental release particulate sample. This fulfills the requirements of performing a monthly composite.

i. After purging seven reactor building volumes, a technical evaluation, prior to reinitiation of a purge following an out of service period, may be conducted in lieu of sampling and analysis.

4.22.2 Dose-Noble Gases

Dose Calculations

Cumulative air dose contributions for the calendar quarter and calendar year shall be determined in accordance with the methodology described in the OFFSITE DOSE CALCULATIONAL MANUAL (ODCM) at least monthly.

Bases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements in Specification 3.18.2 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 are maintained "as low as 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 an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents 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 for Gaseous Effluents (Figure 5.1-3) and are based upon the historical average atmospheric conditions.

4.22.3 Dose-Iodine-131, Iodine-133, Tritium, and Radioactive Materials in Particulate Form.

Dose Calculations

Cumulative dose contributions for the calendar quarter and calendar year period shall be determined in accordance with the methodology described in the (ODCM) OFFSITE DOSE CALCULATION MANUAL at least monthly.

Bases

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 Conditions for Operation in Specification 3.18.3 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 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 an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for estimating doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, and radioactive material in particulate form are dependent on the existing radionuclide pathways to man at or beyond the Site Boundary for Gaseous Effluents (Figure 5.1-3). The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

4.22.4 Gaseous Radwaste Treatment

Surveillance Requirement

Doses due to gaseous releases to areas at and beyond the Site Boundary For Gaseous Effluents (see Figure 5.1-3) shall be projected at least once per 31 days in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM) when Gaseous Radwaste Treatment Systems are not being fully utilized.

The installed VENTILATION EXHAUST TREATMENT SYSTEM and Waste Gas System shall be considered OPERABLE by meeting Specifications 3.18.1, 3.18.2 and 3.18.3.

Bases

The operability of the Waste Gas System and the VENTILATION EXHAUST TREATMENT SYSTEMS ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

4.22.5 Gas Storage Tanks

Surveillance Requirements

The quantity of radioactive material contained in each waste gas decay tank shall be determined to be within the limit in Specification 3.18.5 at least daily when radioactive materials are being added to the tank and the Reactor Coolant System activity exceeds the limits of Specification 3.1.4.

Bases

Restricting the quantity of radioactivity contained in each waste 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 exclusion area boundary (see Figure 5.1-1) will not exceed 500 mrem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

Calculations have shown that the reactor coolant activity must exceed the limits of Specification 3.1.4 before the waste gas decay tank activity approaches the limits of Specification 3.18.5.

4.25 SOLID RADIOACTIVE WASTES

Surveillance Requirements

4.25.1 The solid radwaste systems shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION/DEWATERING to be performed by a contractor in accordance with an approved PROCESS CONTROL PROGRAM.

4.25.2 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of each new mix by testing of at lease one representative test specimen from at least every tenth batch of each type of wet radioactive waste being solidified (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 6.15, to assure SOLIDIFICATION of subsequent batches of waste.

The Process Control Program shall be used to verify the Dewatering of each type of resin and filter media processed to assure established free standing liquid requirements are met.

Bases

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite.

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.

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4.26 RADIOLOGICAL ENVIRONMENTAL MONITORING

Surveillance Requirements

The radiological environmental monitoring samples shall be collected per Table 3.22-1 from the locations shown in the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL and shall be analyzed to the requirements of Tables 3.22-1 and 4.26-1.

Bases

The Radiological Environmental Monitoring Program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby implements Section IV.B.2 of Appendix I to 10CFR50 and 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 ODCM modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. The specified monitoring program is in effect at the present time. Program changes may be initiated based on operational experience and changes in regional population or agricultural practices. The sample locations have been listed in the REMP MANUAL to retain flexibility for making changes as needed.

The detection capabilities required by Table 4.26-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

Table 4.26-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a, d]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Mud and Silt (pCi/kg, wet)
gross beta	4(b)	1×10^{-2}				
^{3}H	2000 (1000(b))					
^{54}Mn	15		130			
^{59}Fe	30		260			
^{58}Co	15		130			150
^{60}Co	15		130			150
^{65}Zn	30		260			
^{95}Zr-Nb	15 (e)					
^{131}I	1 (b)	7×10^{-2}		1	60	
^{134}Cs	15 (10(b))	1×10^{-2}(c)	130	15	60	150
^{137}Cs	18 (10(b))	1×10^{-2}(c)	150	18	80	180
^{140}Ba-La	15 (e)			15 (e)		

Table 4.26-1 (Continued)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a, d]

Table Notation

a. (1) The lower limit of detection (LLD) for a radionuclide presented in this table is the smallest concentration, expressed in picocuries per unit sample, which is required to be detected, if present, in order to achieve compliance with the applicable regulation, given stated operating conditions and calculation methology.

(2) The LLD of a radioanalysis system is that value which will indicate the presence or absence of radioactivity in a sample when the probability of a false positive and of a false negative determination is stated. The probabilities of false positive and false negative are taken as equal at 0.05. The equation for LLD in picocuries per unit sample is given by the equation:

$$LLD = \frac{2.71 + 3.29(Br)^{0.5}}{3.70E4(YEVT)}$$

2.71 = factor to account for Poisson distribution at very low background count rates.

When background is estimated from a blank which has been counted for a specific period, the following applies:

B = estimated background (counts)

$$= b \ \frac{t_s}{t_b}$$

b = blank background (counts)
t_b = blank count time (seconds)
t_s = sample count time (seconds)

$$r = 1 + \frac{t_s}{t_b}$$

Table 4.26-1 (Continued)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a, d]

Table Notation

3.70E4 = disintegrations/second/microcurie

Y = yield of radiochemical process

E = counting efficiency (count/disintegrations)

V = sample volume (liters) or mass (kilograms)

$$T = \frac{[1-\exp(-\lambda t_s)]\exp(-\lambda t_c)}{\lambda}$$

where λ = decay constant (seconds $^{-1}$)
t_c = time from midpoint of collection to start of counting

(3) When spectroscopy is used to analyze the sample, the following LLD equation is used:

$$LLD = \frac{2.71 + 1.65(B)^{0.5}}{3.70E4\ (YEVT)}$$

Where B is the counts in the Region Of Interest.

(4) Analyses shall be performed in such a manner that the stated LLD's 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 LLD's unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

(5) The LLD is defined as a **a priori** (before the fact) limit and not as an **a posteriori** (after the fact) limit.

Table 4.26-1 (Continued)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a, d]

Table Notation

b. LLD for drinking water.

c. LLD shown is for composite analysis. For individual samples, $5x10^{-2}$pCi/m^3 is the LLD.

d. Other peaks which are measurable and identifiable, together with the nuclides in Table 4.26-1, shall be identified and reported.

e. Total for parent and daughter.

4.27 LAND USE CENSUS

Surveillance Requirements

The land use census shall be conducted annually by using methods that will provide the best results, such as door-to-door survey, aerial survey, or by consulting local agriculture authorities.

The land use census or portions thereof, shall be conducted during the appropriate time of the year to provide the best results.

Reports

The results of the land use census shall be included in the Annual Radiological Environmental Operating Report.

Bases

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUAL and ODCM are made if required by the results of this census. 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 requ red to produce the quantity (26 kg/year) of leafy vegetable 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 percent of the garden was used for growing broad-leaf vegetation (i.e.,similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/square meter.

In addition, by gathering information on the liquid effluent pathway and the gaseous effluent pathway, the census will assure that proper radiological environmental monitoring and radioactive effluent controls are in place for the adequate protection of the health and safety of the general public.

4.29 FUEL CYCLE DOSE

Surveillance Requirements

Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.21.2, 4.22.2, and 4.22.3 and in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

Cumulative dose contributions from direct radiation (including outside storage tanks, etc.) shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM). This requirement is applicable only under conditions set forth in the Action Statement of Specification 3.25.

Reports

Special reports shall be submitted as required under Specification 3.25.

Bases

This specification is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the numerical guides for design objective doses of Appendix I or exceeds the reporting levels for the Radiological Environmental Monitoring Program. For the Rancho Seco site, it is unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the plant remains within twice the numerical guides for design objectives of 10 CFR 50, Appendix I and if direct radiation (outside storage tanks, etc.) is kept small. The Special Report will describe a course of action which should result in the limitation of the 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. If the dose to any MEMBER OF THE PUBLIC is evaluated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation which is part of the uranium fuel cycle.

5. DESIGN FEATURES

5.1 SITE

The Rancho Seco reactor is located on the 2,480 acres owned by Sacramento Municipal Utility District, 26 miles north-northeast of Stockton and 25 miles southeast of the City of Sacramento, California. The minimum distance to the boundary of the exclusion area, as defined in 10 CFR 100.3, shall be 2,100 feet.

5.1.1 Exclusion Area

The EXCLUSION AREA shall be shown in Figure 5.1-1.

5.1.2 Low Population Zone

The LOW POPULATION ZONE shall be shown in Figure 5.1-2.

5.1.3 Site Boundary For Gaseous and Liquid Effluents

The SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS for meeting 10 CFR 50, Appendix I guidelines shall be shown in Figure 5.1-3.

5.1.4 Site Boundary For Liquid Effluents

The SITE BOUNDARY FOR LIQUID EFFLUENTS for 10 CFR 20 compliance shall be shown in Figure 5.1-4.

FIGURE 5.1-1

EXCLUSION AREA

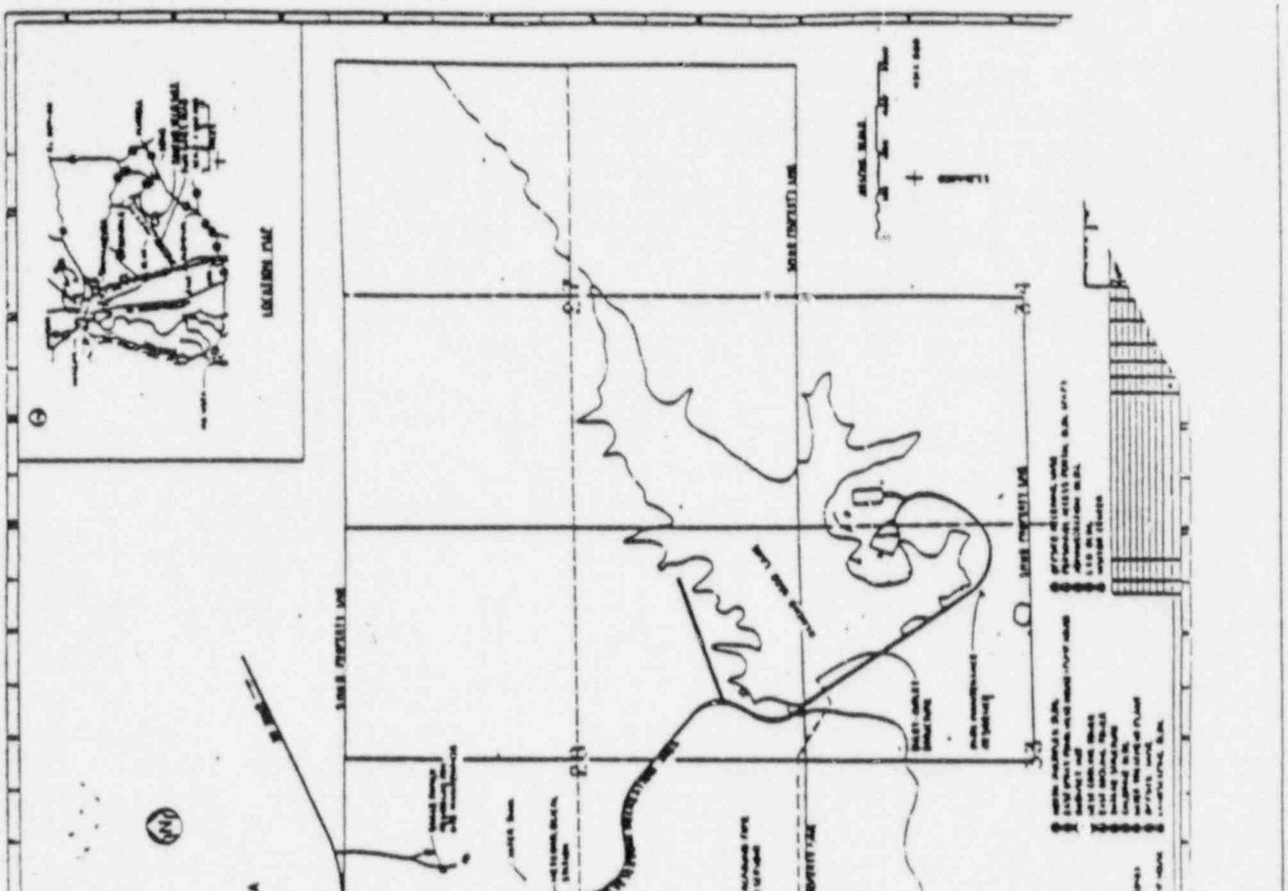


FIGURE 5.1-2

LOW POPULATION ZONE
(5 mile radius)

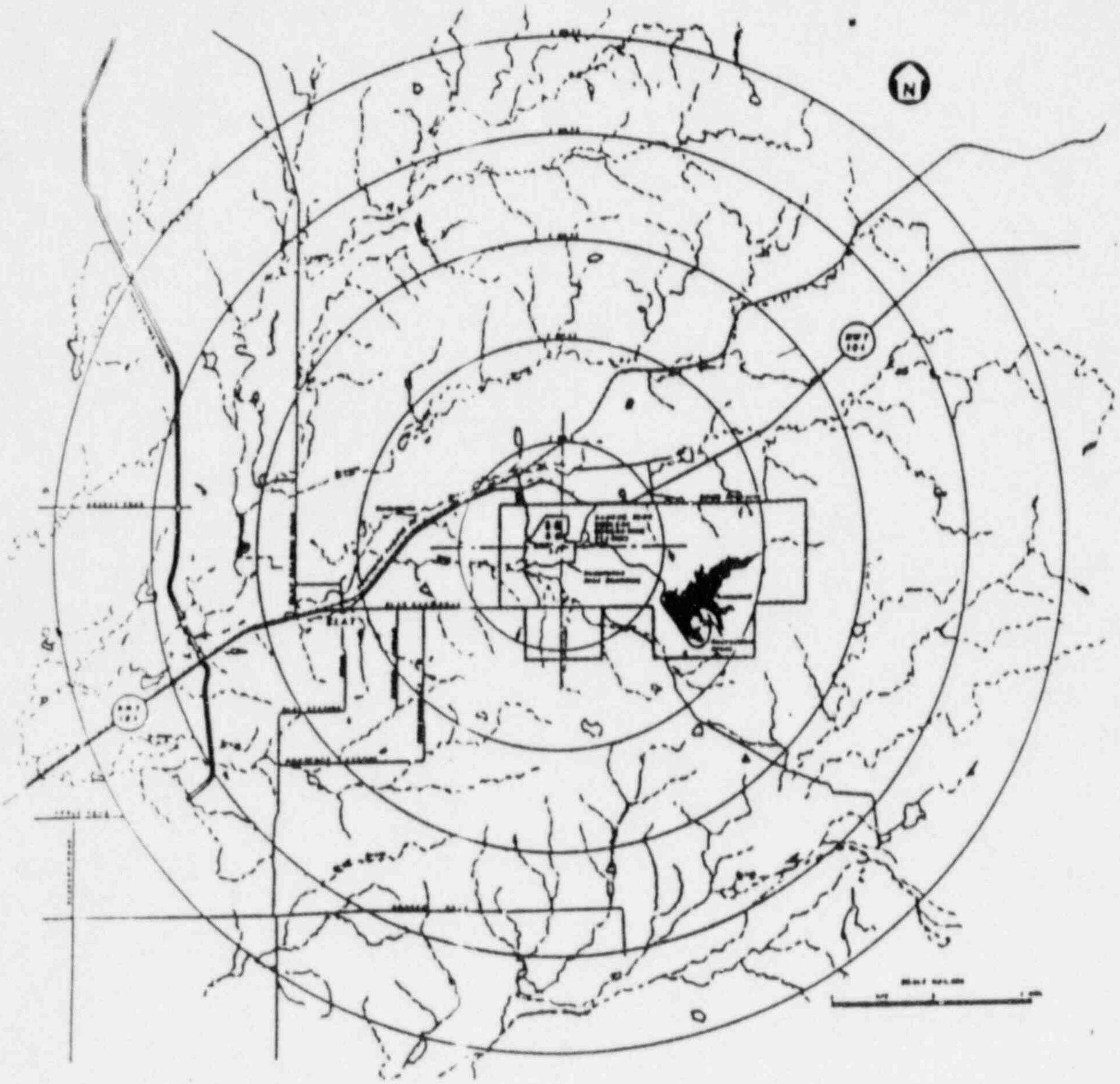


FIGURE 5.1-3

SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS

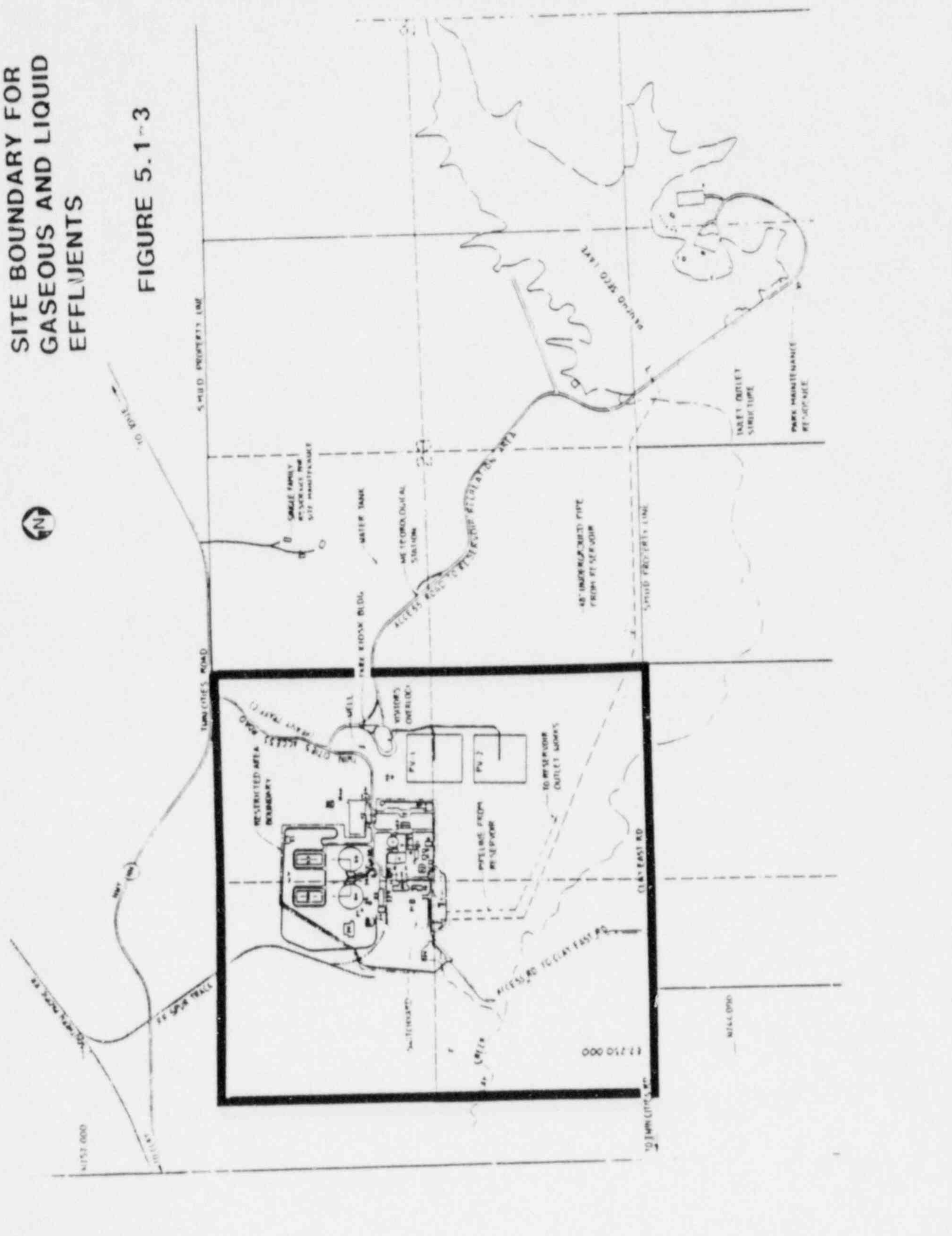
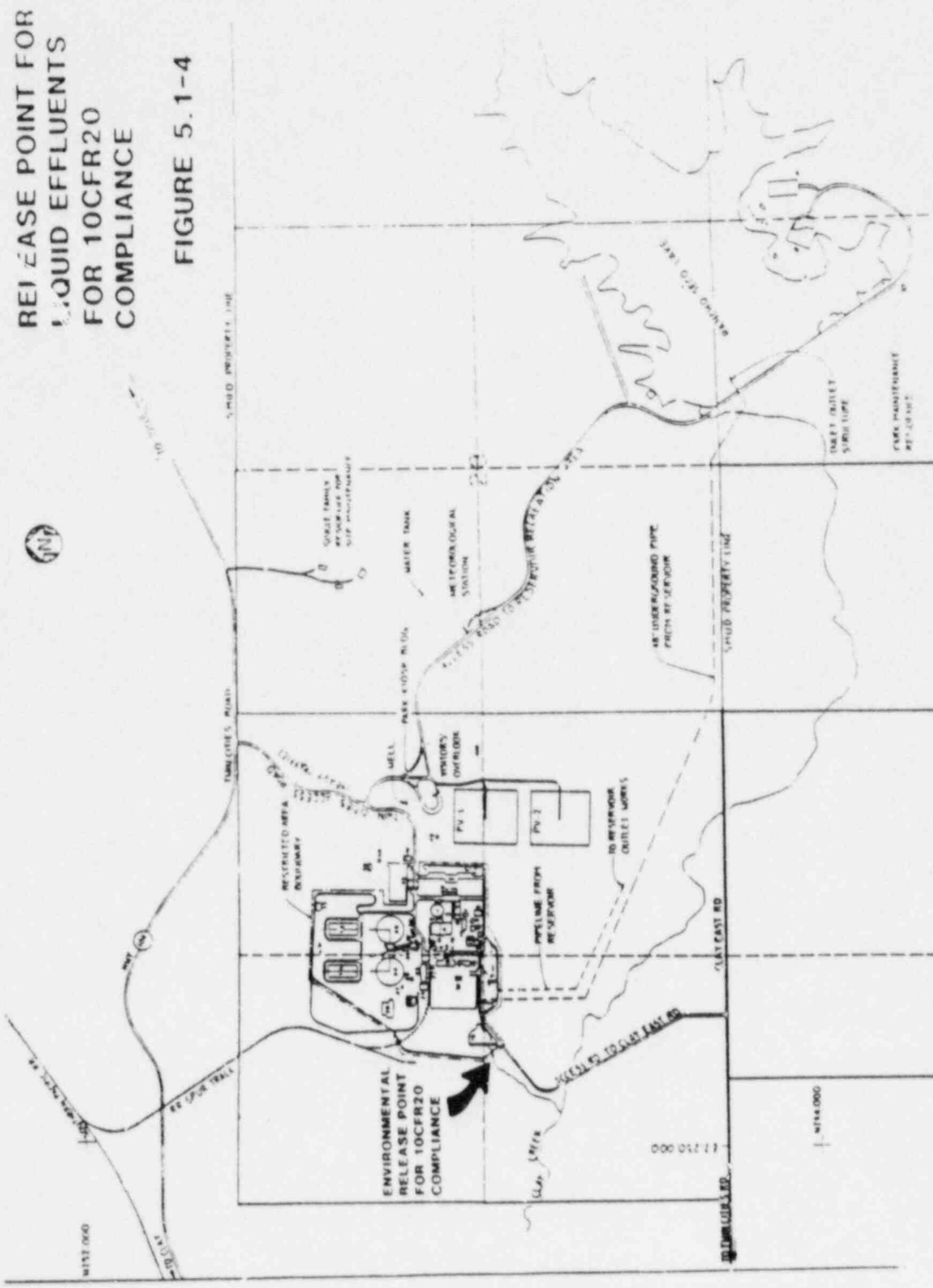


FIGURE 5.1-4

SITE BOUNDARY FOR LIQUID EFFLUENTS FOR 10 CFR 20 COMPLIANCE



RESPONSIBILITIES (Continued)

g. All events requiring a Licensee Event Report as defined by 10CFR50.73 and NUREG 1022 to determine adequacy of corrective action and to detect any degrading trend..

h. Special investigations and reports thereon as requested by the AGM, Nuclear Power Production.

i. The Plant Security Plan and changes thereto.

j. The Emergency Plan and changes thereto.

k. Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL and the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUAL. (See Specifications 6.15 and 6.16.)

l. Major changes to the Radioactive Waste Treatment Systems (Liquid, Gaseous and Solid), and all information required by Specification 6.17.

m. Review of any accidental, unplanned, or uncontrolled 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, and the forwarding of these reports to the Nuclear Plant Manager and to the MSRC.

AUTHORITY

6.5.1.7 The Plant Review Committee shall:

a. Recommend in writing to the AGM, Nuclear Power Production approval or disapproval of items considered under 6.5.1.6(a) through (m) above.

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

c. Provide immediate written notification to the Chairman of the Management Safety Review Committee of disagreement between the PRC and the AGM, Nuclear Power Production; however, the AGM, Nuclear Power Production shall have responsibility for resolution of such disagreements pursuant to 6.5.1.1 above.

RECORDS

6.5.1.8 Records of the PRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each PRC meeting, including appropriate documentation of reviews encompassed by Specification 6.5.1.6e and g, shall be prepared, approved, and forwarded to the AGM, Nuclear Power Production and to the Chairman, Management Safety Review Committee within fourteen (14) days following each meeting.

AUDITS

6.5.4 Audits of facility activities shall be performed under the cognizance of the Director, Quality Assurance, Nuclear. These audits shall encompass:

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

b. The performance, training and qualifications of the District's entire facility technical staff at least once per year.

c. The result of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or methods of operation that affect nuclear safety at least once per six months for those changes not previously audited.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per two (2) years.

e. The Facility Emergency Plan and implementing procedures at least once per two (2) years.

f. The Facility Security Plan and implementing procedures at least once per two (2) years.

g. Any other area of facility operation considered appropriate by the Chief Executive Officer, Nuclear.

h. Compliance with fire protection requirements and implementing procedures at least once per two (2) years.

i. An independent fire protection and loss prevention 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 three (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 processing and packaging of radioactive wastes from liquid systems at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program for Effluent Control and Environmental Monitoring.

6.7 SAFETY LIMIT VIOLATION

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

a. The provisions of 10 CFR 50.36 (c) (1) (i) and 10 CFR 50.72 shall be complied with.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Director, Nuclear Operations and Maintenance, the AGM, Nuclear Power Production, and the Chairman of the MSRC shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRC. 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 MSRC, and the AGM, Nuclear Power Production, within 14 days of the violation.

6.8 PROCEDURES

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.
b. Refueling operations.
c. Surveillance and test activities of safety related equipment.
d. Security Plan implementation.
e. Emergency Plan implementation.
f. Fire Protection Procedures implementation.
g. PROCESS CONTROL PROGRAM implementation.
h. OFFSITE DOSE CALCULATION MANUAL implementation.
i. RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUAL implementation.
j. Quality Assurance Program for the Effluent Control and Environmental Monitoring using the guidance of Regulatory Guide 4.15, Revision 1, February 1979.

6.8.2 Each procedure of 6.8.1 above and changes thereto shall be reviewed and approved as set forth in Specification 6.5.

6.9.2 Radiological Reports

6.9.2.1 Annual Radiological Reports

Annual reports covering the activities of the unit, as described below, for the previous calendar year shall be submitted as follows:

6.9.2.1.1 Annual Occupational Radiation Exposure Report

The Annual Occupational Radiation Exposure Report shall be submitted to the Commission within the first calendar quarter of each calendar year in compliance with 10CFR20.407.

6.9.2.1.2 Annual Exposure Report

The Annual Exposure Report shall be submitted to the Commission within the first calendar quarter of each calendar year in accordance with the guidance contained in Regulatory Guide 1.16.

6.9.2.2 Annual Radiological Environmental Operating Report

6.9.2.2.1 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.2.2.2 The Annual Radiological Environmental Operating Reports shall include summaries and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls (as appropriate). The reports shall also include the results of the Land Use Census required by Specification 3.23. In the event a radionuclide concentration should be confirmed in excess of the reporting level in Table 3.22-2 by environmental measurements, the report shall describe a planned course of corrective action.

6.9.2.2.2 (Continued)

The Annual Radiological Environmental Operating Reports shall include summarized and tabulated results 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; including 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 Interlab Comparison Program. The annual report shall also include information related to Specification 4.29, Uranium Fuel Cycle Dose.

6.9.2.3 <u>Semiannual Radioactive Effluent Release Report</u>

Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.2.3.1 The 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 with data summarized on a quarterly basis.

The Semiannual Radioactive Effluent Release Report shall include a summary of meteorological data collected over the report period. In lieu of submitting all meteorological data with the after July 1 report, the information will be retained in a file onsite and shall be submitted to the NRC upon request.

6.9.2.3.1 Continued)

The Semiannual Radioactive Effluent Release Reports shall include an assessment of the radiation doses from radioactive gaseous and liquid effluents to individuals due to their activities outside the site boundary (Figures 5.1-3 and 5.1-4) during the report period.

The Semiannual Radioactive Effluent Release Reports shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP), RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUAL and OFFSITE DOSE CALCULATION MANUAL (ODCM) pursuant to Specifications 6.15 and 6.16 as well as any major changes to Liquid, Gaseous or Solid Radwater Treatment Systems pursuant to Specification 6.17.

The Semiannual Radioactive Effluent Release Report shall include tables for comparison with Specifications 3.17.2, 3.18.2, and 3.18.3. The July-December report shall include a summary table for comparison with the annual values in Specifications 3.17.2, 3.18.2, and 3.18.3.

The Semiannual Radioactive Effluent Release Report shall also include events described in Specifications 3.17.1, 3.17.3, 3.18.1 and 3.20.

6.9.2.3.1 Continued)

The Semiannual Radioactive Effluent Release Report shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

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

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

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

e. Type of container (e.g., LSA, Type A, Type B, High Integrity), and

f. Solidification agent (e.g., cement).

MONTHLY REPORT

6.9.3 Routine reports of operating statistics, including narrative summary of operating and shutdown experience, of lifts of the Primary System Safety Valves or EMOVs, of major safety related maintenance, and tabulations of facility changes, tests or experiments required pursuant to 10 CFR 50.59(b), shall be submitted on a monthly basis to the Document Control Branch, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Resident Inspector and Regional Office, postmarked no later than the 15th day of each month following the calendar month covered by the report.

LICENSEE EVENT REPORT

6.9.4 The LICENSEE EVENT REPORTS of Specification 6.9.4.1 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC as Licensee Event Reports. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a License Event Report shall be completed and reference shall be made to the original report date, pursuant to the requirements of 10 CFR 50.73.

LICENSEE EVENT REPORT

6.9.4.1 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty (30) days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form, pursuant to 10 CFR 50.73 and the guidance of NUREG-1022.

a. (i) The completion of any nuclear plant shutdown required by the plant's Technical Specification; or

(ii) any operation or condition prohibited by the plant's Technical Specifications; or

(iii) Any deviation from the plant's Technical Specifications authorized pursuant to 10 CFR 50.54(x).

b. Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being:

(i) In an unanalyzed condition that significantly compromised plant safety;

(ii) In a condition that was outside the design basis of the plant; or

(iii) In a condition not covered by the plant's operating and emergency procedures.

c. any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

d. Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported.

e. any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

1. Shut down the reactor and maintain it in a safe shutdown condition;

LICENSEE EVENT REPORT

2. Remove residual heat;

3. Control the release of radioactive material; or

4. Mitigate the consequences of an accident.

f. Events covered in paragraph 6.9.4.1.e of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

g. Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

1. Shut down the reactor and maintain it in a safe shutdown condition;

2. Remove residual heat;

3. Control the release of radioactive material; or

4. Mitigate the consequences of an accident.

h. 1. Any airborne radioactivity release that exceeded 2 times the applicable conc[illegible]ations of the limits specified in Appendix B, Table [illegible] of 10 CFR 20 in unrestricted areas, when averaged over a time period of one hour.

2. Any liquid effluent release that exceeded 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to 10 CFR 20) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour.

i. Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

j. Failure of the pressurizer EMOVs or Primary System Safety Valves.

Special Reports (Continued)

J.	Gaseous and Liquid Radwaste Treatment	30 days (3.18.4 and 3.17.4)
K.	Radiological Environmental Monitoring Program	30 days (3.22)
L.	Deleted	
M.	Solid Radioactive Wastes	30 days (3.21)
N.	Fuel Cycle Dose	30 days (3.25)
O.	Land Use Census	30 days (3.23)
P.	Steam Generator Tube Inspection	30 days (4.17.5)

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

c. Records of facility radiation and contamination surveys.

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

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

f. Records of transient or operational cycles for those facility components designed for a limited number of transients or cycles.

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

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

i. Records of Quality Assurance activities required by the QA 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 PRC and the MSRC.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.14.

m. Records for the Radiological Environmental Monitoring Program.

n. Records of the maintenance of all hydraulic snubbers listed in Table 3.12-1.

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 - Deleted

6.15 PROCESS CONTROL PROGRAM (PCP)

6.15.1 Function

The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured, and that DEWATERING of resin or filter media meets the free standing liquid requirements.

6.15.2 Changes

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

B. Licensee initiated changes to the PCP shall:

1. Be submitted to the Commission by inclusion in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was/were made and 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 or dewatered 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 Plant Review Committee.

2. Become effective upon review and acceptance by the PRC, unless otherwise acted upon by the Commission through written notification to the Licensee.

6.16 OFFSITE DOSE CALCULATION AND RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUALS

6.16.1 Function

6.16.1.1 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall describe the methodology and parameters to be used in the calculation of offsite doses due to the release of radioactive material in gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in various Regulatory Guides as noted in the bases of applicable LCO's.

6.16.1.2 The RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL shall be a manual containing the description of the Rancho Seco Radiological Environmental Monitoring Program. The REMP manual shall contain a description of the environmental samples to be collected, the sample locations, sampling frequencies, and sample analysis criteria.

6.16.2 Any changes to the ODCM or REMP MANUAL shall be made as follows:

A. Licensee-initiated changes:

1. Shall be submitted to the Commission by inclusion in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was/were made and shall contain:

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

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

   c. Documentation of the fact that the change has been reviewed and found acceptable by both the PRC and MSRC.

2. Shall become effective upon a date specified and agreed to by both the PRC and MSRC following their review and acceptance of the change.

6.17 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (LIQUID, GASEOUS, AND SOLID)

6.17.1 Function

The radioactive waste treatment system (liquid, gaseous, and solid) are those systems described in the facility Final Safety Analysis Report or Hazards Summary Report, and amendments thereto, which are used to maintain that control over radioactive materials in gaseous and liquid effluents and in solid waste packaged for offsite shipment required to meet the LCOs set forth in these Specifications.

6.17.2 Major changes to the radioactive waste systems (liquid, gaseous, and solid) shall be made by the following method. For the purpose of this specification, "major changes" is defined in Specification 6.17.3.

Licensee-initiated changes:

1. The Commission shall be informed of all changes by the inclusion of a suitable discussion of each change in the annual USAR update for the period in which the changes were made. 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);

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

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

    d. An evaluation of the change with regard to the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste if different from those previously predicted in the license application and amendments thereto;

    e. An evaluation of the change with regard to the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population if different from those previously estimated in the license application and amendments thereto;

March 17, 1988

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-54

DOCKET NO. 50-312

Replace the following pages of the Appendix "B" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
i	i
20	20
21	21

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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APPENDIX B

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