ATTACHMENT A

MAJOR HIGHLIGHTS OF PROPOSED AMENDMENT NO. 155 REVISION 2 AND REVISION 1

MAJOR HIGHLIGHTS OF PA 155 REV. 2 - RETS

TS 1.0

The following definitions were deleted:

1.27 Radioactive Effluents

The following definitions were modified:

1.15 Offsite Dose Calculation Manual (ODCM)

1.18 Dose Equivalent I-131

1.21 Maximum Exposed (Hypothetical) Individual

The following definitions were added:

1.20 Dewatering

1.27 $\bar{E}$

TS 3.1.4

3.1.4.1 Changed reference from nuclides with half lives longer than 30 minutes to nuclides with half lives longer than 20 minutes.

Changed reactor coolant activity units from μCi/ml to μCi/gm.

TS 3.15

Applicability - During retention basis effluent discharge.

Bases - Stated that the DRCST can be transferred to the A or B RHUT for offsite release. Stated monitors will alarm/trip prior to exceeding the liquid effluent release limits of Specification 3.17.1. Moved reporting requirement to Table 3.15-1 item 1.a.

Table 3.15-1

1.a - Added Retention Basin Effluent Discharge Monitor as environmental release point monitor instead of RHUT monitor. Added requirement to report inoperable (>30 days) liquid effluent instrumentation in next Semi-annual Radioactive Effluent Release Report. This reporting requirement was moved from Bases.

2.a - Clarified action when the RHUT flow measurement device is inoperable.

2.b - Clarified action when the Waste Water flow measurement device is inoperable.

TS 3.16

Applicability - At all times

Bases - Stated monitors will alarm/trip prior to exceeding the gaseous effluent release limits of Specification 3.18.1a.

Table 3.16-1

1.a - Clarified setpoint calculation described in ODCM.

1.d - Clarified action when flow rate device is inoperable.

2.a,f- Combined items 2a. and 2f. because they represent the same monitor. Clarified action statement to address Waste Gas System releases.

3.a - Clarified setpoint calculation described in ODCM.

3.d - Clarified action when flow rate device is inoperable.

* Clarified footnote

TS 3.17

3.17.2 - Changed MAXIMUM HYPOTHETICAL INDIVIDUAL to MAXIMUM EXPOSED INDIVIDUAL. Referenced proper Figure (5.1-3).

3.17.3 - Revised action statement to specify action required if a tank limit is exceeded.

Bases - Deleted a redundant paragraph.

3.17.4 - Established projected 31 day release limits for required use of the Liquid Effluent Radwaste Treatment System. Referenced proper Figure (5.1-3).

TS 3.18

3.18.1

Bases - Gaseous effluent pathways and dose rate to an individual in an unrestricted area are clarified.

3.18.3 - Changed Hypothetical to Exposed

Bases - Added back a statement earlier deleted regarding individuals who may be on site for short periods of time as part of their job.

3.18.4 - Added the 1/4 factor requirement of Standard RETS for gaseous effluents on the operation of Ventilation Exhaust Treatment System.

3.18.5 - Added a new action statement a. to address sampling the online waste gas decay tank when reactor coolant activity reaches the limit of Specification 3.1.4.

TS 3.21

Deleted previous changes. Decided to keep Specification as it presently exists.

TS 3.22

Action b - Added reference to Technical Specification 3.25a, plus referenced controlling release limit specifications 3.17.2, 3.18.2, and 3.18.3.

Action c - Deleted 30 day reporting requirement. If samples become unavailable, replacement sampling locations will be noted in the next annual report.

Table 3.22-1 - Made editorial changes

Table 3.22-2 - No changes. Previous changes supported by a November 1979 NRC Branch Technical Position.

TS 3.23

Placed footnote in proper location

TS 3.25

No changes

TS 3.26

No changes

MAJOR HIGHLIGHTS OF PA 155 REV. 2 - RETS (CONT'D)

TS 4.1

Table 4.1-1	-	Changed item 44c. to 44d. to accomodate an item 44c. added in PA 164.
Table 4.1-3	-	Wording and editorial changes were made. $\bar{E}$ determination was clarified. Table Notes (5) and (6) were added. Existing table note (4) was deleted.
		Waste Gas decay tank, Aux. Bldg. plant vent, and Purge vent deleted from Table. Already addressed in Table 4.22-1.

TS 4.19

Bases	-	Reference to 10 CFR 20.106 was changed to Specification 3.17.1.
Table 4.19-1	-	Table notation was added and proper notations made. The table notation was missing in PA 155, Rev. 1.

TS 4.20

Made editorial and wording changes. Changed reference of 10 CFR 20.106 to Specification 3.18.1.

TS 4.21

Table 4.21-1	-	Changed LLD values to reasonably assure liquid releases will be within 10 CFR 50, Appendix I guidelines.
Table Notation	-	Added note (4) defining LLD as an **a priori** limit.
Tbl. Notation b.		Reworded and clarified the role of the RHUT's and Retention Basins for liquid discharges.
4.21.2	-	Made wording changes. Also added reference to Specification 4.21.2. Referenced proper Figure 5.1-3).
Bases	-	Changed dilution flow rate from 5000 to 8500 gpm during radioactive liquid releases.
Table 4.21-2	-	Changed LLD values to reasonably assure liquid releases will be within 10 CFR 50, Appendix I guidelines.
4.21.4	-	Deleted footnote which gave a Specification implementation condition on system operation.

TS 4.22

Table 4.22-1 - Made editorial and wording changes for clarification.

E. - Deleted I-133 reference to conform with Standard RETS Table Notation b. - Clarified conditions for reactor coolant activity analysis.

4.22.2 - Referenced Specification 3.18.2.

4.22.3 - Referenced Specification 3.18.3.

4.22.4 - No changes

4.22.5 - No changes

TS 4.25

4.25.2 - Made wording changes for clarification. Also included words to address Dewatering.

TS 4.26

Table 4.26-1 - Added note a.(4) to define LLD's as a **a priori** limits. Made editorial changes.

TS 4.29

Made an editorial wording change.

TS 5.0

Changed Figure 5.1-3 and the description of the Figure. The figure now represents Site Boundary for Gaseous and Liquid Effluents for meeting 10 CFR 50 Appendix I Guidelines changed description of Figure 5.1-4. Represents Liquid Effluent boundary for 10 CFR 20 compliance.

TS 6.9.2.2

6.9.2.2.2 - Clarified reporting requirement.

TS 6.9.2.3

6.9.2.3.1 - Stated meteorological data retained in a file on site. Clarified reporting requirement. Deleted reference to Reg. Guide 1.21.

TS 6.9.5

6.9.5.L - Deleted 30 day reporting requirement for substitute monitoring points regarding the REMP.

TS 6.15

6.15.1 - Added reference to Dewater of resins.

6.5.2.B.b - Referenced Dewatering

MAJOR HIGHLIGHTS OF PA 155 - RETS
(Changes through Revision 1)

TS 1.0

The following definitions were added:

1.21 Maximum Hypothetical Individual

1.22 Radiological Environmental Monitoring Program Manual

1.23 Liquid Effluent Radwaste Treatment System

1.24 Ventilation Exhaust Treatment System

1.25 Purge - Purging

1.26 Venting

1.27 Radioactive Effluent

Table 1.9-1 Provides notations for frequencies used in the Technical Specifications

TS 3.15

Action b - Added requirement to report inoperable (> 30 days) liquid effluent instrumentation in next Semi-annual Radioactive Effluent Report

Bases - Stated that the DRCST can be transferred to the A or B RHUT for offsite release

Table 3.15-1

1.a - Added Retention Basin Effluent Discharge Monitor as environmental release point monitor instead of RHUT monitor,

2.a - added RHUT Total Flow Monitor to determine total curies released offsite. Deleted RHUT Discharge Line Monitor (this monitor did not exist)

TS 3.16

Action b - Added requirement to report inoperable (> 30 days) gaseous monitors in the next Semi-annual Radioactive Effluent Release Report

Bases - Deleted reference to non-technical specification Waste Gas Header Monitor

Table 3.16-1

2.a - Deleted reference to automatic termination of release and clarified action statement for Auxiliary Building stack

3. - Renamed "Radwaste Area Vent" "Auxiliary Building Grade Level Vent"

TS 3.17

3.17.1 - Added reference to "Site Boundary for Liquid Effluents" map (Figure 5.1-4)

Action - Add requirement to report exceeding Technical Specification limits

Bases - Clarified that Appendix I guidelines will not be met by this spec

3.17.2 - Referenced Figure 5.1-4

Bases - Added reference to 40 CFR 141 compliance

3.17.3 - Added BWST, DRCST and MWHUT

Bases - Added discussion of required tanks

3.17.4 - Added new specification on Liquid Effluent Radwaste Treatment System (i.e., demin between A & B RHUTS ECN 0775)

TS 3.18

3.18.1 - Referenced Exclusion Area (Figure 5.1-1)

Action - Added reporting requirement when 3.18.1 limits are exceeded

3.18.3 - Referenced Maximum Hypothetical Individual. Referenced I-133 in addition to I-131

3.18.4 - Deleted reduced dose requirement for operability

3.18.5 - Added reporting requirement when 135,000 curie limit is exceeded

TS 3.21

Referenced 10 CFR 20 & 10 CFR 71

Action b - Added reporting requirement

MAJOR HIGHLIGHTS OF PA 155 - RETS (CONT'D)

(Changes through Revision 1)

TS 3.22

Action b - Added reference to Technical Specification 3.25

Bases - Added reference to 40 CFR 141 compliance

Table 3.22-1

3.a - Added composite sample, deleted gross beta and I-131 analysis

3.b - Deleted gross beta and I-131 analysis

3.c - Added ground samples

3.d - Deleted gross beta and included gamma isotopics

4.a - Deleted I-131 analysis, added gamma isotopic

4.b - Added inverterbrater and 2 additional samples, deleted gross beta minus K40 analyses and added gamma isotopic

4.c - Deleted gross beta minus K40 analysis and added gamma isotopic

Table 3.22-2

- Added several reporting levels
- Required analysis to be wet versus dry
- Added reference to 40 CFR 141

TS 3.23

Action a - Added reference to 4.21.2

Action b - Added reporting requirements

TS 3.25

Action - Deleted incorrect reference to TS 3.18.1a and 3.18.1b

TS 3.26

Editorial/Clarification changes only

Table 4.1-1

44.a & b - Separated process and area radiation monitoring changed process surveillance frequencies to Refueling and Quarterly pursuant to standard RETS (need to review surveillance procedures)

TS 4.19

1.a - Add Retention Basin Effluent Discharge Line Monitor, deleted RHUT Monitor (need to review surveillance proceoure)

2.a - Added RHUT Total Flow Monitor (need to review surveillance procedures)

2.b - Added Quarterly Channel Test Pursuant to Standard RETS (need to review surveillance procedures)

TS 4.20

1.a, 2.a, 3.a Revised surveillance frequencies to comply with Standard RETS
1.d, 2.d, 3.d (need to review surveillance procedures)
1.e, 2.e, 3.e

3 - Changed name of "Radwaste Service Area" to "Auxiliary Building Grade Level"

TS 4.21

Table 4.21-1 - Deleted composite analysis
Added H-3 to analysis per batch
Corrected I-131 LLD
Increase dissolved an entrained gas analysis to each batch.
Added definition of LLD in notation 'a'
Deleted notation 'd' on composite
Added notation 'd' on Miscellaneous Water Evaporator

4.21.2 - Added new Dose Tracking Methodology for 10 CFR 50, Appendix I compliance (need to review procedures)

Table 4.21-2 - Added new table of per batch LLDs on selected isotopes and monthly composite analysis.

- Added new bases for 10 CFR 50, Appendix I compliance based on LLDs on CS-137, CS-134, I-131 and H3 equivalent to 50% of Appendix I dose at an estimate annual plant outflow of 20 million gallons and dilution of 5000 gpm.

4.21.3 - Added new surveillance on Liquid Effluent Radwaste Treatment System (need to review surveillance procedures)

MAJOR HIGHLIGHTS OF PA 155 - RETS (CONT'D)
(Changes through Revision 1)

TS 4.22

4.22.1 - Added dose rate due to I-133

Bases - Added reference to Fig. 5.1-3

- Added grass-corn-milk-infant pathway

Table 4.22-1

B - Added note b for gross beta or gamma analysis

C - Changed name of "Radwaste Service Area Vent" to Auxiliary Building Grade Level Vent" added note e on tritium grab samples (need to review surveillance procedures)

D - Added monthly instead of quarterly SR 89/90 analysis for composite sample. Added notes h and g CCD for Noble Gas Monitor reduced to 1 X 10^{-6} pursuant to Standard RETS (need to review surveillance procedures). Decreased Noble Gas LLD to 1 X 10^{-6}.

- Added expanded definition of LLD in Notations

4.22.2 - Administrative/clarification changes only

TS 4.22.3 - Added I-133 in addition to I-131

TS 4.22.4 - Add reference to Figure 5.1-3 and Ventilation Exhaust Treatment Systems

Renumbered spec

TS 4.22.5 - Renumbered spec

Added reference to Fig. 5.1-1

TS 4.23

Deleted see 4.22.4

TS 4.24

Deleted see 4.22.5

TS 4.25

4.25.1 Deleted Reports section, added operability demonstration pursuant to Standard RETS (review procedures)

TS 4.26

Table 4.26-1 Added drinking water requirements. Required wet versus dry analysis, revised LLDs pursuant to Standard RETS, added note (e) on parent daughter totals

Added expanded definition of LLD in Table Notations

TS 4.27

Land use census required during appropriate time of year

TS 4.29

Corrected Technical Specification references

TS 5.0

Added maps of

5.1-1 Exclusion Area

5.1-2 Low Population Zone

5.1-3 Site Boundary for Gaseous Effluents

5.1-4 Site Boundary for Liquid Effluents

TS 6.8

6.5.1.6 - Added the following reviews

- major changes to PCP, ODCM and REMP Manual
- major changes to Radwaste Systems
- any accidental, unplanned or uncontrolled release

6.5.1.7 - Clarification of cross reference

6.5.2.8 - Added audits of the REMP Manual

TS 6.8

6.8.1 - Added REMP Manual implementation

- Committed to Reg. Guide 4.15

TS 6.9

Revised format

6.9.2.3.1 - Clarified required reporting via Semiannual Effluent Report.

6.9.5 - Added special reports for Liquid Radwaste Treatment, Radiological Environmental Monitoring Solid Radioactive Wastes and Land Use Census.

TS 6.15

Clarifications only

TS 6.16

Added REMP Manual

TS 6.17

Clarified expected maximum exposure is to member of public

Report changes in Semiannual Report instead of USAR update

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ATTACHMENT 1

SAFETY ANALYSIS OF PROPOSED AMENDMENT NO. 155 REV. 2

DESCRIPTION:

Proposed Amendment (PA) No. 155 makes comprehensive changes to the Rancho Seco RETS. The changes establish operating boundaries to a greatly improved radiological effluents control program at Rancho Seco. The changes are designed such that the requirements of 10CFR20, 10CFR50, and 40CFR190, and the guidelines of 10CFR50, Appendix I can be met. PA 155 also incorporates to the extent possible Standard RETS of NUREG 0472.

REASON FOR CHANGE:

The changes listed in this Safety Analysis in conjunction with changes to the Offsite Dose Calculation Manual (ODCM), Radiological Environmental Monitoring Program (REMP) and their implementation procedures are being made to establish an adequate radiological effluents control program. The program will allow the District to meet the release requirements and guidelines of 10CFR20, 10CFR50, and 40CFR190 and provide for better management and control of radiological effluents at Rancho Seco.

An NRC staff evaluation of Rancho Seco Radiological Effluent Technical Specification (RETS) and ODCM (US NRC letter to John E. Ward dated 7/22/86) revealed inconsistencies with the Lower Limit of Detection (LLD) values listed in Table 4.21-1 of Rancho Seco Technical Specifications relating to 10CFR50, Appendix I design objectives. The staff evaluation noted radionuclide concentrations near the Rancho Seco site resulting from liquid effluent releases were at levels which could cause the maximum exposed individual to receive doses in excess of 10CFR50, Appendix I guidelines and 40CFR190 limits. Additional NRC citings for nonconformance to 10CFR50, Appendix I were documented in an NRC Region V Inspection Report 86-15 dated 6/6/86. The District's response to the citings on 7/3/86 include commitments to submit Technical Specification revisions which address conformance with the guidelines outlined in 10CFR50, Appendix I.

Another aspect of the changes found in PA 155 is to incorporate where possible the Standard RETS of NUREG 0472 into the Rancho Seco RETS. Many of the changes contained in PA 155 are a direct result of applying the Standard RETS.

EVALUATION AND BASIS FOR SAFETY FINDINGS

The following is a discussion of each Technical Specification change. The changes include administrative changes, changes in accordance with the design basis for Rancho Seco and changes that represent additions or upgrade to existing surveillance requirements and operating limits. The Technical Specification changes listed in this analysis are formatted by individual technical specification changes, accompanied by the existing specification, and a discussion of the change. A safety analysis and a no significant hazards consideration is included with each change. In addition, attached are copies of the Proposed Amendment No. 155 of the Technical Specifications.

1. Existing Specification:

1.13 PROCESS CONTROL PROGRAM

A PROCESS CONTROL PROGRAM (PCP) shall be the manual detailing the program of sampling, analysis, and evaluation within which SOLIDIFICATION of radioactive wastes from liquid system is assured.

1.14 SOLIDIFICATION

Solidification shall be the conversion of liquid radioactive wastes to an immobilized free-standing solid.

New Specification:

1.13 PROCESS CONTROL PROGRAM

PROCESS CONTROL PROGRAM (PCP) - The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION 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).

Discussion:

The changes here are administrative and add clarification to the existing definitions. There is no technical variation in meaning for the Process Control Program or Solidification.

2. Existing Specification:

1.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall be a manual containing the methodology and parameters to be used in the calculation of offsite dose due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring instrumentation alarm/trip setpoints and specific details of the environmental radiological monitoring program.

New Specification:

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.

Discussion:

The details of the radiological environmental monitoring program were originally included in the ODCM. Under these technical specification changes the monitoring program will be a separate document from the ODCM and delineated and titled the Radiological Environmental Monitoring Program (REMP) manual. There is no District commitment to the NRC to separate the ODCM/REMP definitions, but it improves the overall understanding of the District's effluent release program.

2a. Existing Specificat.on:

1.17 SITE BOUNDARY

The boundary of the SMUD property.

New Specification:

1.17 SITE BOUNDARY

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

Discussion:

Figures 5.1-1 through 5.1-4 show the details of the SMUD owned property which form the site boundary. Specifically, the figures show the exclusion area, the low population zone,, the site boundary for gaseous and liquid effluents for meeting 10 CFR 50 Appendix I Guidelines, and the site boundary for liquid effluents for 10 CFR 20 compliance.

3. Existing Specification:

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 as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

New Specification:

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."

Existing Specification:

N/A

New Specification:

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.

Discussion:

The new specification provides clarification of the thyroid dose equivalent pathway of exposure through inhalation. There had been some previous difficulties as to what constitutes "dose equivalent I-131." The changes clarify the definition. TID-14844 is primarily for 10 CFR 100 accident release situations. Reg. Guide 1.109 addresses anticipated releases during normal operations. A definition for Dewatering is added to address dewatering of resins.

4. Existing Specification:

N/A

New Specification

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 radioactive materials in liquid effluents by collecting liquid effluent and providing processing for the purpose of reducing the total radioactivity prior to release to the environment.

1.24 VENTILATION EXHAUST TREATMENT SYSTEM

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

4. New Specification: (Cont.)

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 $\bar{E}$

$\bar{E}$ 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).

Discussion:

The USNRC Regulatory Guide 1.109 is the acceptable methodology for calculating dose resulting from the discharge of radioactive materials in gaseous and liquid effluents for the purposes of comparison with the numerical guides for design objectives of 10CFR50, Appendix I. The definitions of Maximum Exposed Individual (MEI) is necessary to differentiate methodologies for measuring compliance with 10CFR50 Appendix I. The MEI definition is consistent with its use in Reg. Guide 1.109 and 10CFR50, Appendix I dose methodology, and is more conservative than a real dose to a Member of the Public. The definition for the Radiological Environmental Monitoring Program (REMP) manual is provided as a separate document from the ODCM.

The addition of $\bar{E}$ to the definition section represents District conformance to the Standard Technical Specifications and includes the more comprehensive definition found in NUREG-0800. The remaining definitions are added per District commitment to the NRC. They are reflective of the Standard RETS, NUREG-0472.

4a. Existing Specification:

3.1.4 REACTOR COOLANT SYSTEM ACTIVITY

3.1.4.1 The total fission product activity of the reactor coolant due to nuclides with half lives longer than 30 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 left 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 isolated 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 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 release 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 dose at the site boundary will not exceed 0.5 rem, the limit in 10CFR 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 give an upper limit to the potential gamma

4a. <u>Existing Specification:</u> (Cont.)

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:

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

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

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

Where

A = Reactor coolant activity (μCi/ml = Ci/m^3)

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)

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 30 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.

4a. New Specification:

3.1.4 REACTOR COOLANT SYSTEM ACTIVITY

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/E microcuries per gm whenever the reactor is critical. 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 setpoint of the steam safety valves and isolated 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 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 dose at the site boundary will not exceed 0.5 rem, the limit is 10CFR 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

4a. New Specification: (Cont.)

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:

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

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

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

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)

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).

4a. New Specification: (Cont.)

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.

Discussion:

The correct units for reactor coolant activity for the above dose equation is µCi/gm or mCi/Kg.

For a more conservative $\bar{E}$ estimate, radionuclides with half lives greater than 20 minutes shall be used for calculational purposes.

5. Existing Specification:

3.15 RADIOACTIVE LIQUID EFFLUENT 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 are not exceeded.

Applicability During radioactive releases via the pathways identified in Table 3.15-1.

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 are met, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable or change the setpoint so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.15-1.

Bases

During normal operations, all radioactive contaminated water from primary system leaks and drains 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. Only secondary system water is normally released from the plant. The secondary system water, if contaminated, would be released through the Regenerant Hold-Up Tanks.

During periods of primary to secondary leakage, or when the sumps are contaminated, administrative controls require the turbine building sumps liquid effluent to be diverted to the Regenerant Hold-Up Tanks.

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 Regenerant Hold-Up Tanks.

Upon indication of radioactivity in the secondary system, radioactive liquid effluent instrumentation is required to monitor and control, as applicable, the releases of radioactive materials in liquid effluents.

5. Existing Specification: (Cont.)

The alarm/trip setpoints for these instruments shall be calculated in accordance with the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

New Specification:

3.15 RADIOACTIVE LIQUID EFFLUENT MONITORING

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.

5. New Specification: (Cont.)

The secondary system water, if it contains radioactive material is processed through the 'A' or 'B' Regenerant Hold-Up Tanks (RHUTs) to the North or South Retention Basin. The water in the Retention Basin is released offsite 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 turbine building sumps liquid effluent to 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 Hold-Up 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.

Radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology 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.

Discussion

The changes incorporate the provisions documented in the Standard Radiological Effluent Technical Specifications (RETS). Clarification is made that water from the primary system of the plant may be released from the site via the North and South Retention Basins. Additional clarification to the Technical Specifications state that the ODCM contains the methodology to calculate the setpoints of effluent monitoring instrumentation, and not the implementing procedures.

6. Existing Specification:

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. Regenerant Hold-Up Tank Discharge Line Monitor	1	With the monitor inoperable, effluent releases may be resumed provided that prior to initiating a release: 1. At least two independent samples are analyzed in accordance with Specification 3.17. 2. A second member of the facility technical or operational staff will independently verify the release rate calculations and discharge valving. 3. 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 why the inoperability was not corrected in a timely manner.
2. Flow Rate Measurement Devices		
a. Regenerant Hold-Up Tank Discharge Line Monitor	1	With the flow rate measurement device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in situ may be used to estimate flow.

Table 3.15-1 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Flow Rate Measurement Devices (Continued)		
b. Waste Water Flow	1	With the flow rate measurement device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases.

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 devices inoperable, releases to the retention basins may continue provided the total flow is estimated once every 4 hours by a tank level device or pump performance curves.
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.

6. New Specification: (Cont.)

Discussion

The addition of the Retention Basin discharge monitor, which will serve as the District's environmental release point, will provide more effective control of effluent release. The RHUTs A and B total flow monitor is added for measurement of the RHUT volume released to the retention basin and the determination of total offsite dose.

7. Existing Specification:

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 are not exceeded.

Applicability During release via the pathways identified in Table 3.16-1.

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 are met, immediately suspend the release or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.16-1.

Bases

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

The Waste Gas Header Monitor monitors the Waste Gas Holdup System noble gas releases and will provide automatic termination of the release. However, it is located on the system header and monitors the noble gas prior to dilution in the Auxiliary Building ventilation system and passing through HEPA and charcoal filters. The Auxiliary Building Stack alarms and terminates the release automatically if it exceeds the limits. Therefore, as the Auxiliary Building Stack is the effluent release point and will perform the necessary Waste Gas Holdup System release termination, it is listed as the Technical Specification instrument.

7. Existing Specification: (Cont.)

The air ejector exhaust and gland seal exhaust also have individual noble gas monitors. These systems exhaust into the Auxiliary Building ventilation system. Therefore, as the Auxiliary Building Stack is the effluent release point and will alarm if either of these systems release environmentally significant gases, it is used as the Technical Specification instrument.

New Specification:

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 effluent monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

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

7. New Specification: (Cont.)

Bases

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology 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.

Discussion:

The changes represents conformance with the Standard RETS. The gland seal exhaust monitor is being removed per ECN R-1380. Gland seal exhaust is monitored by the Auxiliary Building stack monitor. Additional clarification is made that the methodology to determine the setpoints for the radioactive gaseous effluent instrumentation is contained in the ODCM.

Reference to Specification 3.18 is changed to 3.18.1a in Section 3.16 because 3.16 deals with gaseous effluent monitors as related to dose rate limits which are addressed only by 3.18.1a.

The Waste Gas Header Monitor shall remain in place, but the Auxiliary Building Stack Monitor shall be used to meet the Technical Specification requirement for alarm and automatic termination of release from the Waste Gas System.

8. Existing Specification:

Table 3.16-1

RADIOACTIVE GASES 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 required by Specification 3.16, 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 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 and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.

Table 3.16-1 (Continued)

RADIOACTIVE GASES 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 GASES 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 release	1	With the monitor channel alarm/trip setpoint less conservative than required by Specification 3.16, 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 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 and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.

Table 3.16-1 (Continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack (continued)		
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 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.
f. Waste Gas Holdup System (Auxiliary Building Stack Monitor)	1	With the monitor channel alarm/trip setpoint less conservative than required by Specification 3.16, immediately suspend the release or declare the channel inoperable. With the monitor inoperable, the contents of the tank(s) may be released to the environment provided that prior to initiating the release: a. At least two independent samples of the tank's contents are analyzed, and b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup; Otherwise, suspend release of radioactive effluents via this pathway.

Table 3.16-1 (continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

	Instrument	Minimum Number of Channels Operable	Action
3.	Radwaste Service Area Vent *		
	a. Noble Gas Activity Monitor	1	With the monitor channel alarm/trip setpoint less conservative than required by Specification 3.16, 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 and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.

* The Radwaste Service Area Vent Monitoring System is not yet functional. This specification for this system will become effective when it is declared OPERABLE.

Table 3.16-1 (continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
3. Radwaste Service Area Vent* (continued)		
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken 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.

* The Radwaste Service Area Vent Monitoring System is not yet functional. This specification for this system will become effective when it is declared OPERABLE.

Table 3.16-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent	1	
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 lease 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.**

* 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 (Cont.)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent (continued)	1	
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 (Cont.)

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 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.

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

Table 3.16-1 (Cont.)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack		
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		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.**

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

Table 3.16-1 (Cont.)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack (cont.)		
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. Sample Flow Rate Measuring Devices	1	With the flow rate device inoperable, effluent releases via the pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.
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.

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

Table 3.16-1 (Cont.)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
3. Auxiliary Building Grade Level Vent (cont.)		
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 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 ma; continue provided the flow rate is estimated and recorded at lease once per 4 hours.

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

8. New Specification: (Cont.)

Discussion:

The name for the Radwaste Service Area Vent has been changed to the Auxiliary Building Grade Level Vent. Setpoints for the noble gas activity monitor for the Reactor Building Purge Vent, the Auxiliary Building Grade Level Vent and the Auxiliary Building Stack are based on compliance with 10 CFR 20 requirements as specified in Technical Specification 3.18.1a. Additional changes are based on compliance with Standard RETS, except that the interruption of continuous iodine and particulate sampling is made allowable for periods not to exceed one hour. The "interruption" period is added to allow technicians time to place other continuous monitoring equipment in service once monitor inoperability has been determined.

A note has been added to reference Table 3.5.5-1 for action required because the noble gas activity monitors also function as accident monitors. Additional actions are required for an accident monitor, and these actions are addressed in Table 3.5.5-1.

Items 2a and 2f of Technical Specification Table 3.16-1 have been combined because these two items address the same monitor. This monitor required different actions based on the source of a release, i.e. Waste Gas System vs. all other Auxiliary Building Stack releases.

9. Existing Specification:

3.17 LIQUID EFFLUENTS

3.17.1 CONCENTRATION

The concentration of radioactive material released at any time beyond the site boundary shall be limited to the concentrations specified in 10 CFR 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 $2X10^{-4}$ μCi/ml.

Applicability At all times.

Action

With the concentration of radioactive material released from the site to unrestricted areas exceeding Specification 3.17.1, restore concentration within the specification limits as soon as practicable.

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 will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures within: (1) the Section II.A. Design Objectives of Appendix I, 10 CFR Part 50, to an individual, and (2) the limits of 10 CFR Part 20.106 (e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotopes 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.

New Specification:

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 10CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved

9. New Specification (Cont.)

or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to $2X10^{-4}$ 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 10CFR 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 10CFR 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.

Discussion:

The changes here relate to compliance with the Maximum Permissible Concentration (MPC) of 10CFR20, Appendix B, Table II, Column 2. Addition of Figure 5.1-4 and the action to report in Semiannual Radioactive Effluent Release Report are included for compatibility with the Standard RETS. The existing specification was based on the Standard Radiological Effluent Technical Specifications (RETS) which assumes that for a "standard PWR" compliance with the MPC limit should result in the plant operation being ALARA in terms of the numerical guides for the design objectives of 10CFR50 Appendix I. This is an incorrect assumption for the site specific environmental setting of Rancho Seco, therefore the Appendix I statement in the LCO has been deleted.

10. Existing Specification:

3.17.2 DOSE

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released beyond the site boundary shall be limited:

a. During any calendar quarter 1.5 mrem to the total body and to 5 mrem to any organ; and

b. During any calendar year to 3 mrem to the total body and to 10 mrem to any organ.

Applicability At all times.

Action

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

Bases

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10CFR 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 reasonably achievable." The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculation 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,

10. Existing Specification: (Cont.)

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.

New Specification:

3.17.2 DOSE

The dose or dose commitment to a MAXIMUM EXPOSED (HYPOTHETICAL) INDIVIDUAL from radiological 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 mat als in liquid effluents exceeding any of the above imits, prepare and submit to the Commission within 3 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

10. New Specification (Cont.)

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 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 Release 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.

Discussion:

The changes represent adoption to Standard Technical Specifications NUREG-0472. NUREG-0472 incorporates provisions which include verifying that measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of environment exposure pathways.

The current Rancho Seco Technical Specifications incorrectly references 10CFR50 Appendix I to a real individual (member of the public) where the correct reference is to a Maximum Exposed Individual. Reference to Figure 5.1-3 is added to reflect current site boundary for liquid effluents.

Other wording changes bring the current Technical Specifications into standardization with Standard RETS.

11. Existing Specification:

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. Regenerant Holdup Tanks

b. 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, 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.

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 contents, the concentration at 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. There are two Regenerant Holdup Tanks. The limit applies to each tank individually.

New Specification:

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

11. New Specification: (Cont.)

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 limits, 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.

Discussion

The changes identify all the tank outfalls that will contribute to the Rancho Seco liquid effluent. Assurance is made that the resulting radionuclide concentration in each tank will be less than the limits of 10 CFR 20, Appendix B, Table II, Column 2 so that in the event of an uncontrolled release, the resulting radionuclide concentration in the nearest potable and surface water supply in an unrestricted area will also be less than the aforementioned limits.

12. Existing Specification:

N/A

New Specification:

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 0.60 mrem to the total body or 2.0 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 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

12. New Specification: (Cont.)

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.

Discussion:

This new Technical Specification is added to reflect the addition of LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM (a sluiceable demineralizer system) capable of polishing A & B RHUT contents prior to discharge to the retention basin on an as needed basis. This ensures that the contents of the A & B RHUT can be treated prior to being released to keep the liquid effluent within the dose design objectives set forth in Section II.A of Appendix I of 10 CFR 50.

Because Rancho Seco is a dry site, the Standard Technical Specification 1/4 factor values of 0.06 mrem to the total body and 0.2 mrem to any organ would require all A & B RHUT water be processed through the Sluicable Demineralizers. This requirement would be over restrictive and an unnecessary condition for operation. The values chosen (20% of 10 CFR 50, Appendix I objectives) will ensure ALARA and provide more flexible operations.

13. Existing Specification:

3.18 GASEOUS EFFLUENTS

3.18.1 DOSE RATE

The dose rate at and beyond the site boundary due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

a. The dose rate limit for noble gases shall be 500 mrem/yr to the total body and 3000 mrem/yr to the skin.

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

Applicability At all times

Action

With the dose rate(s) exceeding the above limits, decrease the release rate as soon as practicable to comply with the limit(s) given in Specification 3.18.1.

Bases

This specification is provided to ensure that the dose rate at any time at the site boundary (see Figure 3.18-1) 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 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 outside the restricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individual who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the restricted area 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 restricted area boundary to 500 mrem/yr to the total body or 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 child via the inhalation pathway to less than or equal to 1500 mrem/yr.

13. New Specification:

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 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, 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

13. New Specification: (Cont.)

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.

Discussion:

Changes were made for clarification and to incorporate the Standard RETS.

14. Existing Specification:

3.18.2 NOBLE GASES

The air dose at and beyond the site boundary due to noble gases released in gaseous effluents shall be limited to the following:

a. During any calendar quarter, to 5 mrad for gamma radiation and 10 mrad for beta radiation.

b. During calendar year, to 10 mrad for gamma radiation and 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. This Report will identify the cause(s) for exceeding the limit(s) and define the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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 conform with the guides of Appendix I to 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 will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating

14. Existing Specification: (Cont.)

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 or beyond the restricted area boundary (see Figure 3.18-1) will be based on the historical average atmospheric conditions.

New Specification:

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 above annual limits.

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

14. New Specification: (Cont.)

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.

Discussion:

The changes in this specification incorporate the wording found in the Standard RETS.

15. Existing Specification:

3.18.3 IODINE-131, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

The dose or dose commitment to a member of the public from I-131, from tritium, and from radionuclides in particulate form with half-lives greater than eight days in gaseous effluents released at and beyond the site boundary shall be limited to the following:

a. During any calendar quarter to 7.5 mrem to any organ.

b. During any calendar year to 15 mrem to any organ.

Applicability At all times

Action

With the calculated dose or dose commitment from the release of I-131, tritium, and radionuclides in particulate from with half-lives greater than eight days in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30days a Special Report. This Report will identify the cause(s) for exceeding the limit and define the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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 diffusion factor above that for the restricted area boundary. The ODCM calculational methods for calculating the doses

15. Existing Specification: (Cont.)

due to the actual release rates of the subject materials are required to be consistent with the Methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions.

The release rate specifications for radioiodines and particulates are dependent on the existing radionuclide pathways to man, beyond the site boundary. 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.

New Specification:

3.18.3 DOSE-IODINE-131, IODINE-133, TRITIUM AND RADIOACTIVE MATERIAL 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 effluent 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

15. New Specification: (Cont.)

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.

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 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)

15. New Specification: (Cont.)

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.

Discussion

The changes are in conformance with the Standard RETS based on the dose to the Maximum Exposed Individual. Also, clarification is made as to the site boundary for gaseous effluents. All other changes in this Specification directly reflect wording found in Standard RETS.

16. Existing Specification:

3.19 GASEOUS RADWASTE TREATMENT

The gaseous radwaste treatment system and the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to noble gas releases at and beyond the site boundary (see Figure 3.18-1), would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation over 31 days. The ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site to areas at or beyond the site boundary would exceed 0.3 mrem to any organ over 31 days.

Applicability

Action

a. With a 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 which includes the following information:

1. Explanation of why gaseous radwaste was being discharged without treatment, identification of the equipment or subsystems not OPERABLE and the reasons 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 gaseous radwaste treatment system and the ventilation exhaust treatment system ensures that the systems will be available or 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

16. Existing Specification: (Cont.)

effluents will be kept "as low as is reasonably achievable." The specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

New Specification:

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.

16. New Specification: (Cont.)

Bases

The OPERABILITY of the Waste Gas System and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the system 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.

Discussion:

The changes provide the correct name for the Waste Gas System and provide a reasonable basis for operability. Additionally, the changes are pursuant with the Standard RETS.

17. <u>Existing Specification:</u>

3.20 <u>GAS STORAGE TANKS</u>

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

<u>Applicability</u> At all times

<u>Action</u>

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

<u>Bases</u>

Restricting the quantity of radioactivity contained in each gas storage 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 nearest exclusion area boundary will be 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 cooling 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.

17. New Specification:

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 online 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.

17. New Specification: (Cont.)

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.

Discussion:

Technical Specification section 3.20 was renumbered 3.18.5 and revised in accordance with the guidance provided in the Standard RETS.

The Action Statement is expanded to include reporting requirements specified in Standard RETS. Action Statement a. is maintained because gas storage tank contents are limited to less than 135,000 curies by Technical Specification 3.1.4 as described in the Bases.

18. Existing Specification:

3.21 SOLID RADIOACTIVE WASTES

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

Applicability At all times

Action

With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

Bases

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of Appendix A to 10 CFR 50. The process parameters used in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

New Specification:

N/A

Discussion:

Technical Specification 3.21 is to remain in its existing form. No changes to Specification 3.21 will be made.

19. Existing Specification:

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 Operating Report, 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, or to malfunction of automatic sampling equipment. If the latter, efforts shall be made to complete corrective action prior to the end of the next sampling period).

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, 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 limits to be exceeded. This report will define corrective actions to reduce emissions such that potential annual exposures will meet the Specifications 3.17.2, 3.18.2, and 3.18.3. The exceeding of Table 3.22-2 levels may result from more than one radionuclide in the sampling medium if:

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

Dose calculations will include all measured radionuclides of plant origin. 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.

19. Existing Specification: (Cont.)

c. With milk or fresh leafy vegetation samples unavailable from any of the sample locations required by Table 3.22-1, prepare and submit to the Commission within 30 days a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. 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 environmental monitoring program as replacement locations, if available.

Bases

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The 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 ODCM to retain flexibility for making changes as needed.

With no drinking water intakes downstream of the plant, surface water and runoff water samples do not have to meet drinking water requirements and sample frequencies.

New Specification:

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

19. New Specification: (Cont.)

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:

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

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 Specification 3.17.2, 3.18.2, and 3.18.3. This report is not required if the measures 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

19. New Specification: (Cont.)

provided the locations from which the replacement samples were obtained are added to the Radiological Environmental Monitoring Program as replacement locations, if available.

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.

Discussion

The text changes to the Action statements are editorial and clarification changes. Specifically,

1. "Environmental" inserted between "Radiological" and "monitoring" to differentiate between onsite protection and offsite environmental monitoring.

2. "Reporting" inserted between "exceeding the" and "level" in item b. to clarify the content of Table 3.22-2.

The test changes to the Bases statements are similar to those in the Action statements. The sample locations shall be defined in the REMP manual and not the ODCM.

19. New Specification: (Cont.)

The changes represent conformance with the Standard RETS to provide reporting levels for radionuclide concentrations in the environmental samples (Table 3.22-2) in order to appropriately identify when concentrations of radioactive materials and levels of radiation may be higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. In addition, the Radiological Environmental Monitoring Program (REMP) will account for all potential land, water usage, and food radiological exposure pathways that exist downstream from Rancho Seco. The sampling and collection frequency (Table 3.22-1) will allow determination of long-term buildup of concentrations of radionuclides in bottom sediment, doses due to ingesting aquatic foods (bottom feeding fish) and direct radiation from long-term buildup of radionuclides on land irrigated with contaminated water.

Table 3.22-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
A. Radioiodine and Particulates	8	Continuous operation of sampler 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 radioactivity greater than or equal to 24 hours following filter change. Perform gamma isotopic analysis on each sample where gross beta activity is greater than 10 times the appropriate control samples for the same sample period. Perform gamma isotopic analysis on composite (by location) sample at least once per quarter.
2. DIRECT RADIATION	Greater than 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 ODCM.

Table 3.22-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	3	Grab sample collected monthly.	Gross Beta and I-131 analysis of each suspended and dissolved fraction. Tritium analysis at least once per quarter.
b. Runoff	1	Grab sample collected fortnightly.	Gross Beta and I-131 analysis of each suspended and dissolved fraction. Tritium analysis at least once per quarter, plus gamma isotopic analysis on dissolved and suspended fractions.
c. Mud and Silt	2	At least once semi-annually. One pint sample of the top 3" of material 2 ft. from shoreline.	Gross Beta on each sample.

* Sample locations are shown in the ODCM.

Table 3.22-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	4	At least once per fortnight when animals are on pasture; at least once per month at other times.	I-131 analysis of each sample.**
b. Fish	1	At least semi-annually. One sample of each of several species as shown in the ODCM.	Gross Beta minus K-40 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 ODCM.	Gross Beta minus K-40 analysis on edible portion of each sample.**

*Sample locations are shown in the ODCM.
**Gamma Isotopic Analysis when Table 3.22-2 levels are exceeded.

20. New Specification:

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 radioactivity at least 24 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 Inverte-brates	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.

*Technical Specification sample locations are identified in the REMP MANUAL.

20. New Specification: (Cont.)

Discussion

The changes made in Table 3.22-1 support additional sampling and collection frequency of the liquid effluent pathway documented in the REMP manual. The NRC considered that the District was deficient in this aspect of the Rancho Seco radiological effluent monitoring program. Also, the Table 3.22-1 changes reflect conformance to the Standard RETS by the addition of a monthly composite sample and the deletion of Gross Beta and I-131 analysis for the waterborne surface exposure pathway.

21. Existing Specification:

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

Limiting Conditions for Operation

Table 3.22-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

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

Table 3.22-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	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		2			

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

21. New Specification: (Cont.)

Discussion

Table 3.22-2 values represent a reporting level of radioactive concentrations in environmental samples and not a regulatory limit. The requirement for exceeding the reporting level is to submit a report that evaluates the effluent observations against the EPA regulations documented in 40 CFR 190 (Tech Spec 3.25).

Additional reporting levels are included for cesium and iodine (I-131) analysis in fish and food products which are in the liquid effluent pathway. Food products include meat pathways and are identified in the REMP Manual. Requirements for gross beta were deleted for food and fish pathways. This is in conformance with the Standard RETS.

22. Existing Specification:

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 vegetables in each of the 16 meteorological sectors within a distance of five miles.

Applicability At all times

Action

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.22.3, identify the new locations in the next Semiannual Radioactive Effluent Release 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% 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 environment monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s) (via the same exposure pathway) may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted. Identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

Bases

This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program 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

22. Existing Specification: (Cont.)

to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 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% 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.

New Specification:

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 (REMP) MANUAL can be kept current with the existing environmental and societal uses surrounding Rancho Seco.

* 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.

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

22. New Specification: (Cont.)

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 Reporting and also include in the report a revised figure(s) and table for the REMP manual reflecting the new location(s).

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 vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine 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 of 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.

Discussion

The Rancho Seco REMP, utilizing the guidance of NUREG-0472, provides for an annual land use census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modification to the

22. New Specification: (Cont.)

monitoring program are made if required by the results of the census. The changes here will include an addition of liquid pathway surveillance so that existing environmental and societal uses of land surrounding Rancho Seco can be kept current. Identification of gardens in the summer, rather than the middle of winter will be included in the census to assure a more realistic sampling of gardens. In addition, liquid and gaseous pathways are identified and reportable as land use census dose results which will be included in the Annual Radiological Environmental Operating Report.

23. Existing Specification:

3.25 FUEL CYCLE DOSE

The annual dose or dose commitment to a member of the public due to releases of radioactivity and radiation from uranium fuel cycle sources is limited to ≤25 mrem to the total body or any organ (except the thyroid, which is limited to ≤75 mrem).

Applicability At all times

Action

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, 3.18.1.b, 3.18.2.a, 3.18.2,b, 3.18.3.a, or 3.18.3.b, calculations should be made to determine whether the above limits of Specification 3.25 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405(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, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose(s) exceed the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the 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. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from the plant radioactive effluents exceed twice the design objective doses of Appendix I. For the Rancho Seco site it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the

23. Existing Specification: (Cont.)

plant remains within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC 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 estimated 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 190.1 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 nuclear fuel cycle.

New Specification:

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) over 12 consecutive months.

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

23. New Specification: (Cont.)

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, over 12 consecutive months 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 in 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 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 purpose 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

23. New Specification: (Cont.)

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.

Discussion

Changes here represent conformance to the Standard RETS. The dose limits of 40 CFR 190 have been incorporated into 10 CFR 20. 40 CFR 190 is now the controlling limitation for Fuel Cycle Dose.

24. Existing Specification:

3.26 INTERLABORATORY COMPARISON PROGRAM

The contractor performing the analysis of radiological environmental program 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.

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 samples are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

New Specification:

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.

24. New Specification: (Cont.)

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.

Discussion:

This specification has been revised to clarify and to reference the specification requiring submittal of the Annual Radiological Environmental Operating Report.

25. <u>Existing Specification:</u>

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 and area radiation monitoring systems	W	M	Q	
b. 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. Auxiliary Feedwater Start Circuit				
a. Phase Imbalance/Underpower RCP	S	NA	R	
b. Low Main Feedwater Pressure	NA	M	R	
49. Pressurizer Water Level	M	NA	R	
50. Auxiliary Feedwater Flow Rate	M	NA	R	
51. Reactor Coolant System Subcooling Margin Monitor	M	NA	R	
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	
56. Safety Valve Position Indicator (Backup Detector) Acoustic	M	NA	R	

25. New Specification:

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	
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	Q	(1) Battery check
48. Auxiliary Feedwater Start Circuit				
a. Phase imbalance/Under-power RCP	S	NA	R	
b. Low Main Feedwater Pressure	NA	M	R	
49. Pressurizer Water Level	M	NA	R	
50. Auxiliary Feedwater Flow Rate	M	NA	R	
51. Reactor Coolant System Subcooling Margin Monitor	M	NA	R	
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	
56. Safety Valve Position Indicator (Backup Detector) Acoustic	M	NA	R	

Discussion:

Item 44.a has been split into two items and given surveillance requirements in accordance with the guidance of Standard RETS.

Item 44c. has been relabled 44d. to allow for the inclusion of item 44c. established in PA 164.

25. Existing Specification:

TABLE 4.1-3
MIMIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor coolant	a. Radio-chemical analysis(1) Ē determination(5) b. Gross activity(1) (3) c. Tritium radioactivity d. Chemistry (Cl and O_2) e. Boron concentration f. Fluoride	M Semiannually 3/week M 3/week 2/week M
2. Borated water storage tank water sample	Boron concentration	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) b.Iodine analysis(2)(3)	Weekly Weekly
6. Concentrated boric acid tank	Boron concentration	2/week and after each makeup
7. Waste gas decay tank	Isotope analysis	Q, Prior to release
8. Auxiliary Building plant vent	Isotope analysis (4)	W
9. Spray additive tank	NaOH concentration (3)	Q and after each makeup
10. Purge vent	Gross activity	During each Purge
11. Blowdown from cooling towers	Gross activity (3)	M

(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)When activity level exceeds ten percent of the limits of 10CFR20, the sampling frequency shall be increased to a minimum of once each day.

(5)Ē determination will be started when gross beta-gamma activity analysis indicates greater than 10μCi/ml and will be redetermined each 10μCi/ml increase in gross beta-gamma activity analysis. A radio chemical analysis for this purpose shall consist of a quantitative measurement of 95% of radionuclides in reactor coolant with half lives of >30 minutes.

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

25. New Specification:

Item		Check	Frequency
1.	Reactor coolant	a. Radio-chemical analysis(1) $\bar{E}$ determination (3)(4)(6) b. Gross activity(1) (3) c. Tritium radioactivity d. Chemistry (Cl and O_2) e. Boron concentration f. Fluoride	M Semiannually 3/week M 3/week 2/week 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) b. Iodine analysis(2)(3)	Weekly 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
8.	Cooling Tower water	Gross activity (3)	M

25. New Specification: (Cont.)

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.

Discussion:

Changes to Table 4.1-3 represent conformance to Standard Technical Specifications of NUREG-0800.

Footnotes (5) and (6) were added to accommodate system shutdown for maintenance and to ensure reactor stability for radio-chemical analysis and $\bar{E}$ determination at power, respectively.

Cooling tower blowdown is the same as cooling tower water. $\bar{E}$ activity is measured in μCi/gm not μci/ml.

Waste Gas Decay Tank, Auxiliary Building Plant Vent, and Purge Vent sampling frequencies are addressed in Table 4.22-1 and should not be in Table 4.1-.3

The existing specification footnote (4), regarding 10% of 10 CFR 20 limits, is deleted because it represents the old concentration release rate requirement. Specification 3.18.1a is now the controlling requirement.

26. Existing Specification:

4.19 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

Surveillance Requirements

The maximum setpoint shall be determined in accordance with procedures as described in the ODCM and shall be recorded on the release permits.

Each radioactive liquid effluent monitoring instrumentation channels 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 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 are met.

Bases

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

New Specification:

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 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.

26. New Specification: (Cont.)

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.

Discussion

The changes represent conformance to the Standard Radiological Environmental Technical Specifications (RETS). Also reflected is that the ODCM contains the methodology to ensure that the setpoints are adequate for the alarm/trip to occur prior to exceeding the limits of Specification 3.17.1.

27. Existing Specification:

Table 4.19-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation				
a. Regenerant Hold-Up Tank Discharge Line Monitor	D (1)	M (5)	R (2)	Q (3)
2. Flow Rate Monitors				
a. Waste Water Flow	D (4)	NA	R	NA

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 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.

(5) During periods of known activity in the regenerant tank, perform a source check daily during releases via this pathway.

27. New Specification.

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 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.

27. New Specification: (Cont.)

Discussion

New monitors are added on the Retention Basin Effluent, which is now the current Rancho Seco environmental control point. The changes to the table notation are pursuant with Standard RETS.

Clarification is made to indicate that a totalizer is used to measure total flow downstream of the dilution flow.

4.20 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

The maximum setpoints shall be determined in accordance with procedures as described in the 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 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 are met.

Bases

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

The flow rates in the Reactor Building Purge Vent, Auxiliary Building Stack and Radwaste Service Area Vent are constant as they use single speed fans. The Reactor Building Purge Vent has two different flow rates, winter and summer, however administrative controls assure using the correct flow rate where applicable. The actual flow rate of the ventilation systems are periodically determined by surveillance procedures. The flow rate measurement devices are used only as flow indicating devices and not for actual measurement of flow rate. Also, as these flow rate devices must be removed from the ventilation system 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.

28. New Specification:

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 10CFR 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 flow rate 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.

28. <u>New Specification</u>: (Cont.)

<u>Discussion</u>:

Clarification is made that provides that the ODCM describes the methodology to establish the setpoints to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR20.106. The "Radwaste Service Area Vent"names has been editorially changed to "Auxiliary Building Grade Level Vent."

29. Existing Specification:

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(1)	M	Q(2)	Q(3)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device	W	NA	BY	A
e. Sampler Monitor Flow Rate Measurement Device	W	NA	BY	A
2. Auxiliary Building Stack				
a. Noble Gas Activity Monitor	D(1)	M	Q(2)	Q(3)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device*	W	NA	BY	A
e. Sampler Monitor Flow Rate Measurement Device	W	NA	BY	A

* This flow rate device is not yet installed. This specification for this system will become effective when it is declared OPERABLE.

Table 4.20-1 (Continued)

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
3. Radwaste Service Area*				
a. Noble Gas Activity Monitor	D(1)	M	Q(2)	Q(4)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device	W	NA	BY	A
e. Sampler Monitor Flow Rate Measurement Device	W	NA	BY	A

* The Radwaste Service Area Monitoring System is not yet functional. The specification for this system will become effective when it is declared OPERABLE.

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 termination 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 not set in operate mode.

Table 4.20-1 (continued)
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

(4) The Channel Test shall also demonstrate that 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 not set in operate mode.

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 isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

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

Discussion:

The changes to Table 4.20-1 are made to be pursuant with the Standard Radiological Environmental Technical Specifications. Table Notation (6) is added to ensure safety of technicians performing the channel test on the Auxiliary Building stack flow-rate device due to the device's location.

30. Existing Specification:

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 values given in Specification 3.17.1.

The radioactivity content of each batch of radioactive liquid waste 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 ODCM to assure that the concentration at the point of release is limited to the values in Specification 3.17.1.

Post-release analyses of samples from batch releases shall be performed in accordance with Table 4.21-1. The results of the post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release are limited to the values in 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 will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures within: (1) the Section II.A Design Objectives of Appendix I, 10 CFR Part 50, to an individual, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for 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 methods described in International Commission on Radiological Protection (ICRP) Publication 2.

30. Existing Specification: (Cont.)

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

New Specification:

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 material released in liquid waste effluent 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 any equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

30. New Specification (Cont.)

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.

Discussion:

The changes here represent compliance with 10 CFR 20 Appendix B, Table II, Column 2, which does not guarantee that a licensee will meet the numerical guides for design objectives of 10 CFR 50, Appendix I.

The surveillance requirements relate to the concentration LCO (3.17.1). The bases for the LCO is in the Standard RETS which assumes that for a "Standard PWR" compliance with the maximum permissible concentration (MPC) limit should result in the plant operation being ALARA in terms of the numerical guides for the design objectives of 10 CFR 50, Appendix I. This assumption is incorrect for the site specific environmental setting of Rancho Seco, therefore reference to 10 CFR 50, Appendix I compliance is removed from the bases statement in surveillance standards.

Table 4.21-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Lim Of Detect (LLD) (uCi/ml)(
A. Batch Waste Release Tanks(b)	Each Batch P	Each Batch P	Mn-54, Fe-59, Co-58, Co-60 Zn-65, Mo-99, Cs-134, Cs-137 Ce-141, and Ce-144 (c)	5×10^{-7}
			I-131	1×10^{-6}
	One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Each Batch P	M Composite(d)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Each Batch P	Q Composite(d)	Sr-89, Sr-90	5×10^{-8}

Table 4.21-1 (Continued)

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

a. The lower limit of detection (LLD) is defined in the ODCM.

b. A batch release is the discharge of liquid wastes of discrete volume. Prior to sampling, each batch will be isolated and thoroughly mixed, to assure representative sampling.

c. Other peaks which are measureable and identifiable, together with the listed nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analysis should not be reported as being present at the LLD level.

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.

31. New Specification:

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

31. New Specification: (Cont.)

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 B, 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.

31. New Specification: (Cont.)

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 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 nuclides: 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 measurable and identifiable, together with with 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.

31. New Specification: (Cont.)

Discussion

Table notation items are clarified for wording. A new set of LLDs are established at a concentration equivalent of about 50% of the values in Technical Specification 3.17.2, except for H-3, which is at a concentration equivalent of less than 1%. These LLDs ensure the capability of complying with 10 CFR 20 and also provide the capability for reasonable assurance of compliance with 10 CFR 50, Appendix I dose guidelines. (Refer to Attachment 3 for added detail on LLD changes). Composite analysis for 10 CFR 50, Appendix I compliance has been moved to a new Table 4.21-2. Table 4.21-2 includes gross alpha and Sr-90. Added conservatism was also included by increasing the sampling and minimum analysis of dissolved and entrained gases (gamma emitters).

32. Existing Specification

4.21.2 Doses

Dose Calculations

Cumulative dose contributions from liquid effluents shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM) at least monthly.

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 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 is to 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.

New Specification:

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,

32. New Specification (Cont.)

b. Monthly, based on gamma-emitter and tritium analyses of 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 calculation 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 material 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

32. New Specification (Cont.)

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 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 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 10CFR50, 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 40 CFR 141.

Discussion:

The proposed amendment addresses the NRC concern in the NRC Inspection Report 86-15 which states the Rancho Seco LLD values are in error since the use of these values can result in releases in excess of the limits of 40 CFR 190 and the design objectives of 10 CFR 50, Appendix I. The details of the dose tracking surveillance and new dose methodology is described in the ODCM implementing procedures. The tracking system contains review and approval mechanisms at multiple management levels based on the pre-release dose calculations. The addition of the LLD values in Tables 4.21-1 and 4.21-2 are specified in Attachment 3 of this safety evaluation. Historical data was used in determining the LLD values and indicated that Cs-134, Cs-137, I-131 and tritium could be used as indicators for all nuclides in the Rancho Seco effluent mix.

Pre-Restart procedure modifications are being implemented to increase the minimum average dilution flow rate during a radioactive liquid discharge to 8,500 gpm.

32. New Specification (Cont.)

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

32. New Specification: (Cont.)

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.

32. New Specification: (Cont.)

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 RHUT's, each batch will be isolated, and then thoroughly mixed, to assure representative sampling. Samples will also be collected from the Retention Basins 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, Ce-141, Ce-144, and Ba-140. Other peaks which are measurable 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.

Discussion:

A definition of LLD has been added to the Table notation of Table 4.21-2 as detailed in Attachment 3 of this Safety Analysis. Table notation items are clarified for wording. New LLDs are established at a concentration equivalent to about 50% of the 10 CFR 50 Appendix I release guidelines for prebatch release sampling and 10% of the 10 CFR 50 Appendix I release guidelines for post release composite sampling, except for H-3, which is at a concentration equivalent to less than 1%. These LLDs provide the capability for reasonable assurance of compliance with 10CFR50, Appendix I dose guidelines.

Composite analysis for 10CFR50, Appendix I is now in new Table 4.21-2. Added conservatism was also included by increasing the sampling and minimum analysis of dissolved and entrained gases (gamma emitters).

Prior to discharge from the 'A' or 'B' RHUT, samples are collected and analyzed for accountability of activity in the Retention Basins.

33. Existing Specification:

4.21.3 LIQUID HOLDUP TANKS*

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 tans provides assurance that in the event of an uncontrolled release of the contents, the concentration at the nearest surface water supply in an unrestricted area would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2.

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*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.

New Specification:

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 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.

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*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding 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.

33. New Specification: (Cont.)

Discussion

Addition is made to the bases that makes assurance that potable water is to be within the 10 CFR 20 Appendix B concentration limits for radioactive materials in the event of an uncontrolled liquid holdup tank release.

34. Existing Specification:

N/A

New Specification:

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 reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

Discussion

The new surveillance requirement is added per District commitment to the NRC, assuring the operability of the radwaste treatment system in the event of liquid effluent release.

4.22 GASEOUS EFFLUENTS

4.22.1 Dose Rate

Surveillance Requirements

The release rate of noble gases in gaseous effluents shall be controlled by the offsite dose rate as established in Specification 3.18.1. The dose rate shall be determined in accordance with the ODCM.

The noble gas effluent continous monitors, as listed in Table 3.16-1, shall use monitor setpoints to limit offsite doses within the values established in Specification 3.18.1.

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 at and beyond the site boundary, due to Iodine-131, tritium, and all radionuclides in particulate form with half-lives greater than 8 days released in gaseous effluents, shall be determined to be within the required limits by using the results of the sampling and analysis program, specified in Table 4.22-1, in performing the calculations of dose rate beyond the site boundary in accordance with the ODCM.

Bases

This specification is provided to ensure that the dose rate at any time at the site boundary 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 outside the restricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For 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 diffusion factor above that for the restricted area 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 restricted area boundary to 500 mrem/year to the total body or 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 a child via the inhalation pathway to less than or equal to 1,500 mrem/year.

35. New Specification:

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 effluent, 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 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 10CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10CFR 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 10CFR Part 20 (10CFR 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

35. Existing Specification: (Cont.)

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	Each Tank Grab Sample P	Each Tank P	Principal Gamma Emitters (f)	1×10^{-4}
B. Containment Purge	Each Purge Grab Sample(e) P	Each Purge P	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
C. Auxiliary Building Stack, and Radwaste Service Area Vent	M(b,c) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as listed in A,B,C above	Continuous	W(d) Charcoal Sample	I-131	1×10^{-12}
	Continuous	W(d) Particulate Sample	Principal Gamma Emitters (f) (I-131, Others)	1 x 10-[illegible]
	Continuous	M Composite Particulate Sample	Gross Alpha	1 x 10-[illegible]
	Continuous	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble Gas Monitor	Noble Gases Beta or Gamma (Gross)	1 x 10-[illegible] as Xe-133

Table 4.22-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

a. The lower limit of detection (LLD) is defined in the ODCM.

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

d. Samples shall be changed at least weekly with analyses to 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 LLD's may be increased by a factor of 10.

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

f. Principle gamma emitters for which the LLD applies are: Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-135m for gaseous samples and Mn-54, Fe-59, Co-58, Co-60, 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 reported; other peaks that are measurable and identifiable will also be reported.

New Specification:

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

36. New Specification:

Table 4.22-1 (Continued)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Table Notation

155→ 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)

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

36. New Specification: (Cont.)

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 = 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 analysis 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.

36. New Specification: (Cont.)

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 may be 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.

Discussion:

Clarification is made to note f. in relation to reporting of measured LLD values. Note i. was added to allow more operational flexibility for Reactor Building purges. Other wording changes were made to reflect Standard RETS.

Specification 4.22.1 is modified to allow monitor setpoints to demonstrate compliance with Specification 3.18.1a. Calculations to assess compliance are required only when a monitor exceeds its alarm setpoint or it is inoperable.

37. Existing Specification:

4.22.2 NOBLE GASES

Dose Calculations

Cumulative air dose contributions for the quarterly or yearly period as applicable shall be determined in accordance with 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 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 maintained "as low as reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conform with the guides of Appendix I to 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 will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary will be based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

New Specification:

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.

37. New Specification: (Cont.)

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.

Discussion

The changes here represent conformance with the Standard RETS. Clarification is made for the frequency of determination of the cumulative air dose contributions.

4.22.3 Iodine-131, Tritium and Radionuclides in Particulate Form

Dose Calculations

Cumulative dose contributions for the quarterly or yearly period as applicable shall be determined in accordance with the ODCM 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 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 approved by the NRC for calculating the doses due to the actual release rates of the subject materials are required to be consistent with the methodology provided in Regulatory Guide 1.109, "Calculating of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive material in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, beyond the site boundary. The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

38. New Specification:

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, 10CFR 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 10CFR 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.

38. New Specification: (Cont.)

Discussion:

The changes here represent conformance with Standard RETS. Clarification is made for the frequency of determination of the cumulative air dose contributions.

39. Existing Specification:

4.23 GASEOUS RADWASTE TREATMENT

Dose Projections

Doses due to gaseous releases beyond the site boundary shall be projected at least monthly in accordance with the ODCM.

Bases

The operability of the gaseous radwaste treatment 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 these 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 Design Objective Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

New Specification:

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

39. New Specification: (Cont.)

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 50.36, 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.

Discussion

These changes are pursuant with the Standard RETS. Additional clarification is made for surveillance standards.

40. Existing Specification:

4.24 GAS STORAGE TANKS

Surveillance Requirements

The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limit of 3.20 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 gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest site boundary will not exceed 500 mrem. This is consistent with Standard Review Plant 15.7.1, "Waste Gas System Failure."

New Specification:

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.

40. New Specification: (Cont.)

Discussion

Clarification is made that the exclusion area boundary is the limiting boundary for determining dose to an individual from an uncontrolled release from the gas storage tanks.

4.25 SOLID RADIOACTIVE WASTES

Surveillance Requirements

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

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

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.15, to assure SOLIDIFICATION of subsequent batches of waste.

Reports

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, Large Quantity), and

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

4.25 (Continued)

Bases

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.

41. New Specification:

4.25 SOLID RADIOACTIVE WASTES

4.25.1

Surveillance Requirements

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

Surveillance Requirements

The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of each new mix by testing at least 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 solution).

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.

41. New Specification: (Cont.)

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 10CFR Part 50.26 and General Design Criterion 60 of Appendix A to 10CFR 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.

Discussion:

Changes here represent the addition of periodic operability demonstrations for the solid radwaste system as recommended in the Standard RETS. The REPORT section of the existing specification has been deleted and is being added to Technical Specification 6.9.2.3.1 via this Proposed Amendment.

42. Existing Specification:

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 ODCM and shall be analyzed to the requirements of Tables 3.22-1 and 4.26-1.

Bases

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The 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 ODCM to retain flexibility for making changes as needed.

New Specification:

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 10 CFR 50 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 pathway.

42. New Specification (Cont.)

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.

Discussion:

Changes here represent that the REMP manual will be the governing document for the radiological environmental surveillance requirements. Reference to 40 CFR 141 is pursuant to the Standard RETS.

43. Existing Specification:

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

Surveillance Standards

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/gm, dry)	Milk (pCi/l)	Food Products (pCi/gm, dry)	Mud and Silt (pCi/gm, dry)
gross beta	4[(b)]	1×10^{-2}	1×10^{-1}		1×10^{-1}	2×10^{-1}
^{3}H	2000 (1000[(b)])					
^{54}Mn	15		3×10^{-2}			
^{59}Fe	30		3×10^{-2}			
^{58}Co	15		2×10^{-2}			
^{60}Co	15		6×10^{-2}			
^{65}Zn	30		6×10^{-2}			
^{95}Zr-Nb	15					
^{131}I	1	7×10^{-2}		1	3×10^{-2}	
^{134}Cs	15	1×10^{-2}[c]	8×10^{-2}	15	8×10^{-2}	
^{137}Cs	18	1×10^{-2}[c]	6×10^{-2}	18	6×10^{-2}	
^{140}Ba-La	15			15		

43. New Specification:

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

Surveillance Standards

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)		

Discussion:

Changes to Table 4.26-1 represent District conformance to the Standard RETS. Lower maximum LLD values are defined for Cesium detection in drinking water and milk. In addition, LLD values are established for the liquid effluent pathway in mud and silt for Cesium and cobalt detection.

The addition of establishing LLD values for MUD and silt addresses the NRC concern in the July 22, 1986 staff evaluation that the mud silt effluent pathway model did not take into account long-term buildup of concentrations of radionuclides in bottom sediments, thereby imparting doese to ingesting aquatic foods (bottom-feeding fish).

The values for the LLD's come from a November 1979 NRC Branch Technical Position.

44. Existing Specification:

Table 4.26-1

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

Table Notation

a. The LLD is defined in the ODCM.

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.

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.

44. Existing Specification: (Cont.)

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)

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

44. Existing Specification: (Cont.)

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:

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

Where B is the counts in the Region Of Int' est.

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

Table 4.26-1 (Continued)

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

Table Notation

(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.

b. LLD for drinking water.

c. LLD shown is for composite analysis. For individual samples, 5×10^{-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.

Discussion:

Changes to Table 4.26-1 represent District conformance to the Standard RETS. Lower maximum LLD values are defined for Cesium detection in the drinking water and milk. In addition, LLD values are established for the liquid effluent pathway in mud and silt for Cesium and cobalt detection.

The addition of establishing LLD values for mud and silt addressed the NRC concern in the July 22, 1986 staff evaluation that the mud and silt effluent pathway model did not take into account long-term buildup of concentrations of radionuclides in bottom sediments, thereby imparting doses to ingesting aquatic foods (bottom-feeding fish).

45. Existing Specification:

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.

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 areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program 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 required 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% 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.

New Specification:

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.

45. New Specification (Cont.)

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 required 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% 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 adequate protection of the health and safety of the general public.

Discussion:

The changes represent the surveillance requirements as it relates to the LCO in Tech Spec 3.23 Land Use Census. Additions include liquid effluent pathway surveillances for current land use. Identification of the REMP as the environmental monitoring vehicle is included in the text of the Bases.

46. Existing Specification:

4.29 FUEL CYCLE DOSE

Surveillance Requirements

Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.17.2.a, 3.17.2.b, 3.18.1.a, 3.18.1.b, 3.18.2.a, 3.18.2.b, 3.18.3.a, and 3.18.3.b, and in accordance with the Offsite Dose Calculation Manual (ODCM).

Reports

Special reports shall be submitted as required under Specification 3.25.

Bases

This specification is provided to meet the dose limitations of 40CFR190. The specification requires the preparation and submittal of a Special Reports whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For the Rancho Seco site, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40CFR1.90 if the plant remains within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of the annual dose to a member of the public to within the 40CFR190 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 estimated to exceed the requirements of 40CFR190, the Special Report with a request for a variance (provided the release conditions resulting in violation for 40CFR190 have not already been corrected), in accordance with the provisions of 40CFR190.11, is considered to be a timely request and fulfills the requirements of 40CFR190 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 nuclear fuel cycle.

46. Existing Specification:

New Specification:

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 40CFR190 that have been incorporated into 10CFR20 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 40CFR190 if the plant remains within twice the numerical guides for design objectives of 10CFR50, 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 40CFR190 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 40CFR190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40CFR190 have not already been corrected), in accordance with the provisions of 40CFR190, is considered to be a timely request and fulfills the requirements of 40CFR190 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.

Discussion:

The modifications are in conformance with the Standard RETS.

47. Existing Specification:

5. DESIGN FEATURES

5.1 SITE

Specification

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. FSAR figure 1.1-2 shows the plan of the site. The minimum distance to the boundary of the exclusion area, as defined in 10 CFR 100.3, shall be 2,100 feet. (1), (2)

REFERENCES

(1) FSAR paragraph 1.2.1

(2) FSAR paragraph 2.2.1

New Specification:

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.

47. New Specification: (Cont.)

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.

Discussion:

The addition of site schematic drawings identifying the affected areas of effluent release is pursuant with the Standard RETS.

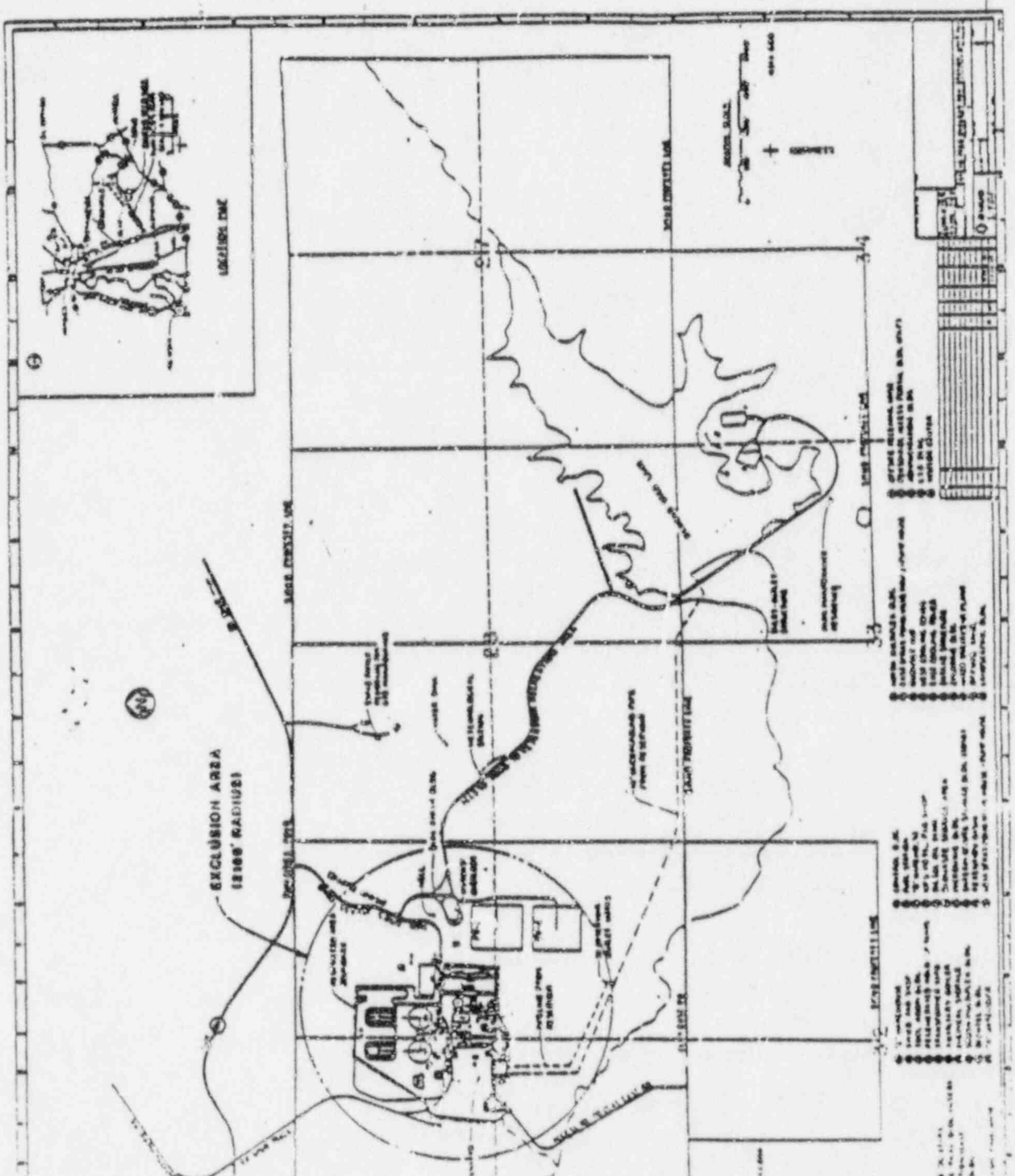
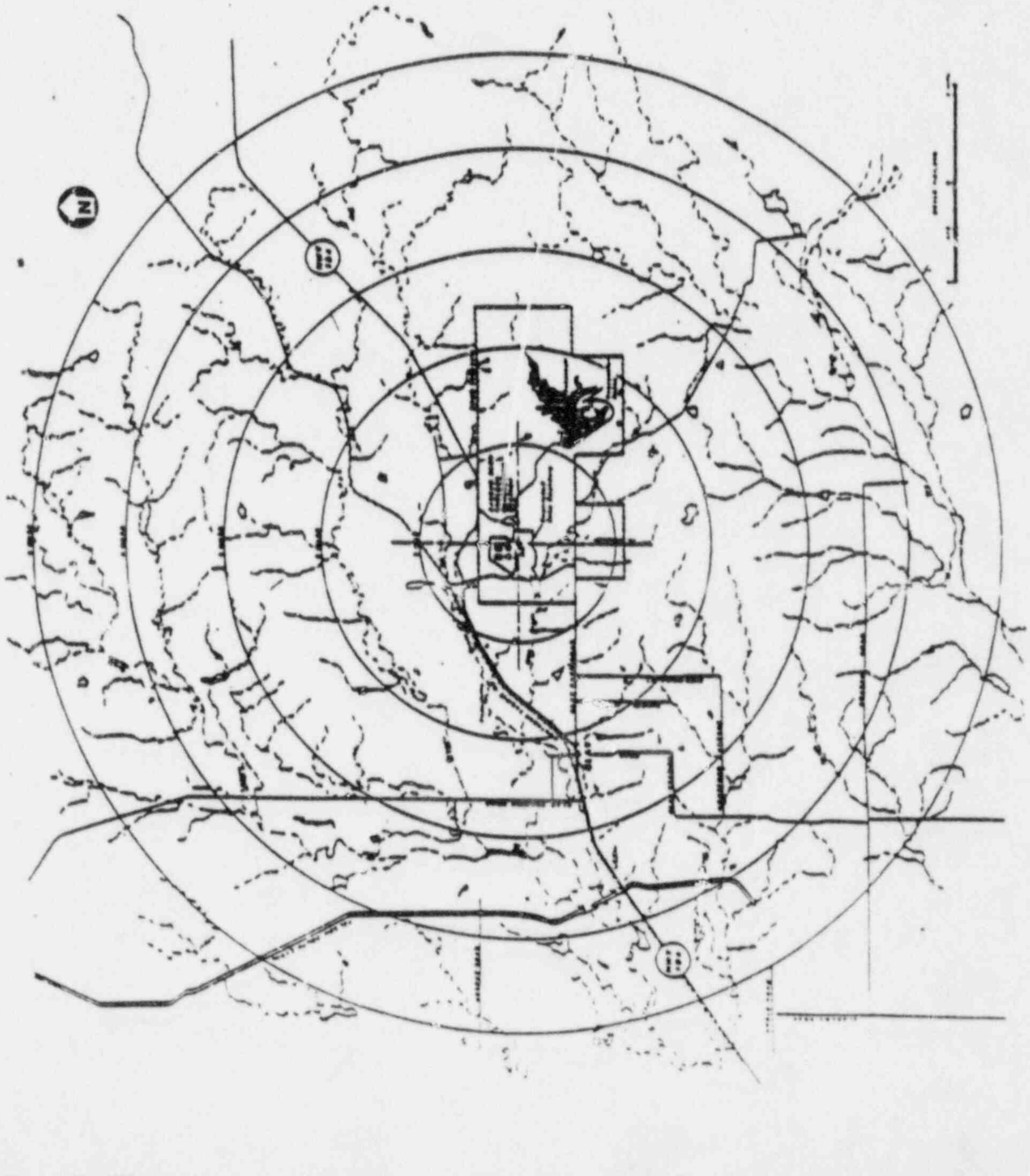


FIGURE 5.1-1 EXCLUSION AREA

FIGURE 5.1-2
LOW POPULATION ZONE
(5 mile radius)



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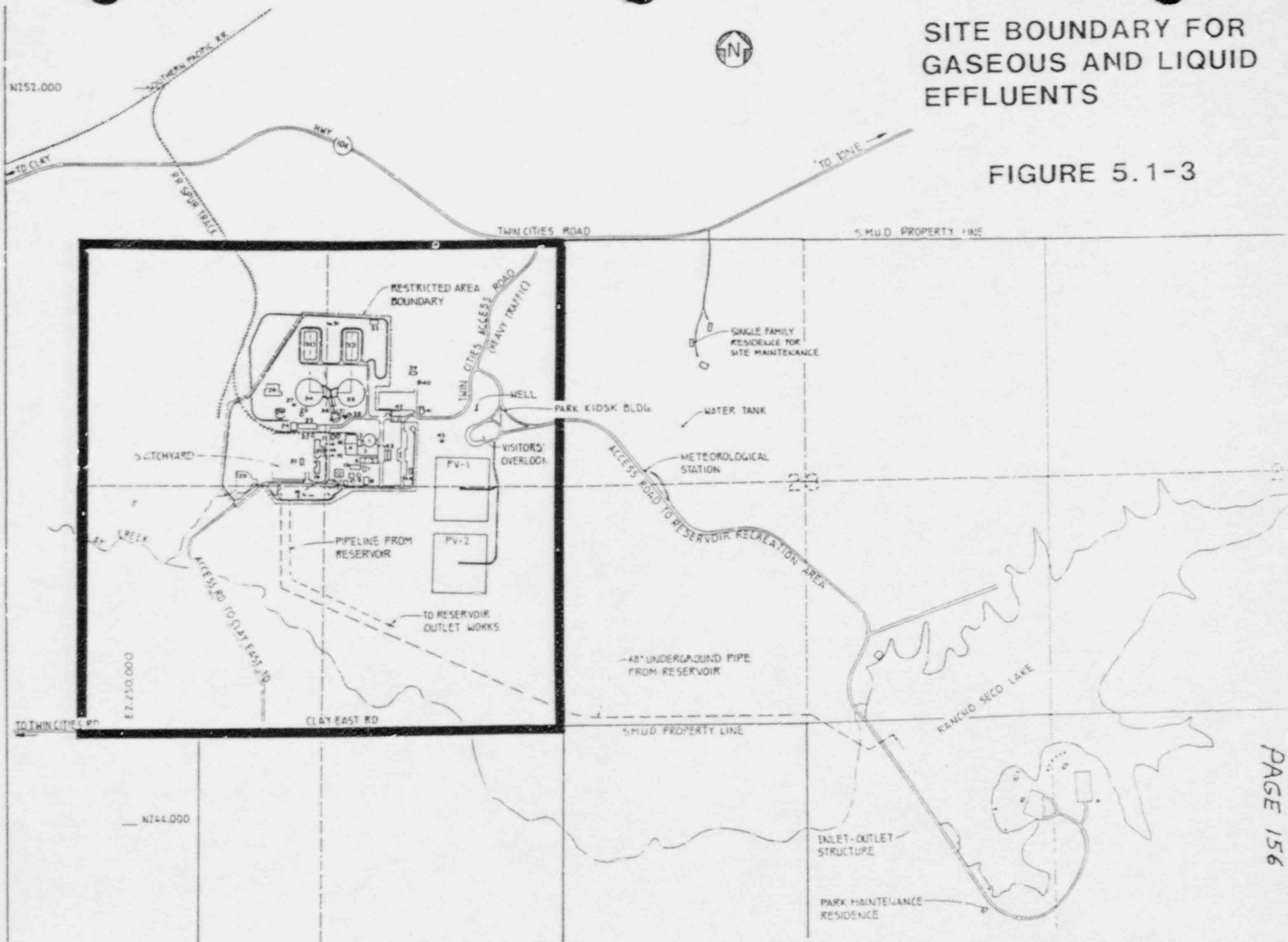
SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS

FIGURE 5.1-3



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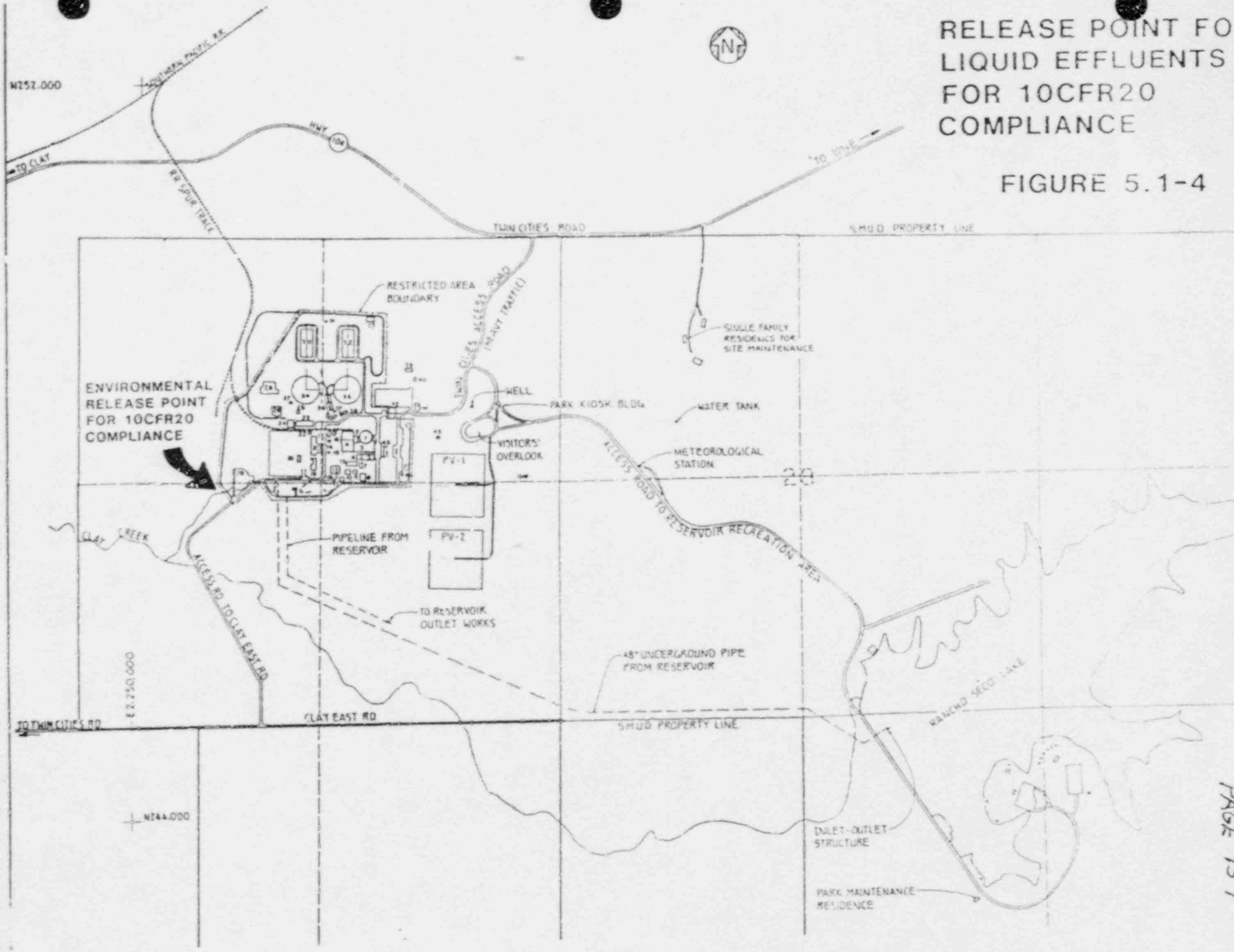

RELEASE POINT FOR LIQUID EFFLUENTS FOR 10CFR20 COMPLIANCE

FIGURE 5.1-4

48. Administrative And Editorial Changes:

See Attachment 2 to this safety analysis which is a subset of the overall Technical Specification change of Proposed Amendment No. 155. The changes are made to the Table of Contents and Figures and Chapter 6, Administrative Controls.

Safety Analysis For Technical Specification Changes (Non-Administrative/Editorial) Items 1 - 47

The changes to the Technical Specifications will address the NRC staffs' review undertaken in connection with contamination found in the vicinity of the Rancho Seco plant. Technical Specification inconsistencies were found between the Lower Limit of Detection (LLD) as listed in Table 4.21-1 of the Technical Specification Surveillance Standards (Section 4.21.2) and the Technical Specification Sections 3.17.2 and 4.21.2. The NRC position stated that because of the highly atypical characteristics of the Rancho Seco cooling water system and of the receiving waters, liquid effluent releases with the current LLD valves, could result in excess of 10 CFR 50, Appendix I and limits specified in 40 CFR 190.

These concerns are addressed in this safety analysis in the following groupings for adoption to the Standard Radiological Effluent Technical Specifications (RETS) of NUREG-0472:

1) Concentration Limits
2) Dose Limits
3) Radiological Environmental Monitoring Program
4) Land Use Census
5) Reporting, Procedures and Audits
6) Instrumentation

1. Concentration

NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," provides calculational models for dose contributions for implementing 10 CFR Part 50, Appendix I "as low as is reasonably achievable" requirements. For liquid effluent releases, a near field average dilution factor is used which takes into account the maximum undiluted liquid waste flow, the combined liquid releases for each unit, and the mixing effects in the receiving water body in the near field of the discharge structure. For plants with non-recirculating main condenser cooling systems, the mixing effects in the receiving water body are ignored for conservatism. However, for plants with recirculating cooling systems, where cooling water discharge flow rates are much less than for plants with

Safety Analysis For Technical Specification Changes (Non-Administrative/Editorial (Cont.)

1. Concentration

non-recirculating cooling systems, credit is allowed for mixing effects in the near field of the receiving water body up to the degree of non-recirculating cooling water.

Rancho Seco has a recirculating main condenser cooling system. Based on a comparison on Environmental Statements for various nuclear power plants, the Rancho Seco design average discharge flow rate is one of the lowest of all U. S. nuclear power plants. As with similar plants, the liquid waste discharge includes condenser cooling and service water system blowdown, and other minor streams in addition to liquid radwaste effluents. However, atypically, at Rancho Seco there is little or no dilution of liquid wastes after discharge from the plant discharge structure due to the almost total absence of a receiving water body comprised of water other than from the plant discharge. Consequently, no credit is provided in the Rancho Seco ODCM for mixing in the receiving water body in the near field of the discharge structure.

The existing specification for concentration in the Rancho Seco Technical Specifications (RSTS) was based on the Standard Radiological Effluent Tech Specs (RETS) NUREG-0472, which assumed that for a "standard PWR" compliance with the Maximum Permissible Concentration (MPC) on an hour by hour release basis will result in the plant being ALARA. This assumption was incorrect due to the fact that Rancho Seco is a "dry" plant for effluent releases, i.e., little dilution of liquid wastes after discharge due to the absence of a receiving water body. To correct the inconsistency, changes are made to the RSTS (3.17.1) to state that Rancho Seco will comply with the Maximum Permissable Concentration of 10 CFR 20, Appendix B, Table II, Column 2. There is no guarantee that the design objectives for 10 CFR 50, Appendix I, which is dose based, can be met singularly using concentration based LLD's, therefore Appendix I compliance statements are removed from the LCO and surveillance bases. Surveillance requirements (Tech Spec 4.21.1) for radionuclide concentrations in liquid effluent releases have been revised to reflect additional conservatisms in the radioactive liquid waste sampling and analysis program by increasing the sampling and analysis frequency of dissolved and entrained gases with newly established LLD's (see Table 4.21-1). The historical mix of radionuclides released at Rancho Seco provided the basis for establishing that Cs-134 and Cs-137 are the major dose indicators for all gamma emitters, except iodines. The setting of the LLD's for Cs-134 and Cs-137 at a concentration equivalent of 50% of Technical

1. Concentration (Cont.)

Specification values has been developed and included in this amendment (see Attachment 3 Bases for Lower Limit of Detection Values for Rancho Seco Liquid Effluents).

The Offsite Dose Calculation Manual (ODCM) will address dose related contributions from liquid effluents, thereby, addressing ALARA. The Offsite Dose Calculation Manual (ODCM) will be the governing document for meeting concentration standards of radioactive effluent releases.

Basis For No Significant Hazards Determination

The proposed changes to the Technical Specifications regarding the concentration of radionuclides in liquid effluents do not involve a significant hazards consideration because operation of Rancho Seco in accordance with these changes would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. The proposed changes do not affect plant design or alter the safety/accident analysis of Chapters 11 and 14 of the Updated Safety Analysis Report (USAR). The changes provide clarification and ensure that plant liquid effluents comply with the requirements of 10 CFR 20, Appendix B, Table II, Column 2 regarding the maximum permissible concentration of radioactive materials released from the site to areas beyond the site boundary for liquid effluents. Therefore, these changes do not significantly increase the probability or consequences of an accident previously evaluated;

(2) create the possibility of a new or different kind of accident from any previously analyzed. These proposed changes reflect clarification and compliance with the requirement of 10 CFR 20, Appendix B, Table II, Column 2 and do not create the possibility of a new or different kind of accident from any previously analyzed;

(3) involve a significant reduction in a margin of safety. These changes ensure compliance with the requirements of 10 CFR 20, Appendix B, Table II, Column 2 and 20 CFR 20.106 regarding the maximum permissible concentration of radioactive material in the liquid effluents and the resultant exposure limit to a member of the public. Compliance with the ALARA guidelines of 10 CFR 50, Appendix I is assured by the dose limits and dose tracking methodology of the changes to Technical Specifications 3.17.2 and 4.21.2. The revision of Tables 4.21-1 and 4.21-2 reflect additional conservatism in the radioactive liquid waste sampling and analysis program. The revision of definition 1.15 reflects the restructuring of the Offsite Dose Calculation Manual. Implementing procedures for dose calculation will be issued in

1. Concentration (Cont.)

separate documents. Environmental monitoring has been removed from the ODCM, incorporated in the Radiological Environmental Monitoring Program (REMP) Manual, and added as an additional Technical Specification definition. Therefore, these revisions do not involve a significant reduction in a margin of safety.

2. Dose

The existing Rancho Seco Technical Specification 3.17.2, which is provided to implement the requirements of 10 CFR Part 50, Appendix I, requires that the annual dose to a member of the public from radioactive materials in liquid effluents be limited to 3 millirems to the total body and to 10 millirems to any organ. Rancho Seco Technical Specification 3.25, which is provided to meet the dose limitations of 40 CFR 190, requires that the annual dose to a member of the public due to releases of radioactivity and radiation from fuel cycle sources be limited to 25 millirems to the total body or any organ (except the thyroid, which is limited to 75 millirems).

The NRC position stated that incorporation of the model RETS LLD values in the Rancho Seco RETS was in error since use of these values can result in releases of radioactive materials to which offsite doses may be attributed (through the use of the methodology of the Rancho Seco ODCM) that are in excess of the limits provided by the RETS to implement the regulations, 10 CFR Part 50, Appendix I, and 40 CFR Part 190.

The District has developed a position with respect to the use of Lower Limits of Detection for compliance with 10 CFR Part 50, Appendix I. In this position, two sets of LLD's are used: the first is based on the capabilities of the Rancho Seco Chemistry facilities; the second is based on the capabilities of environmental-level facilities.

The historical mix of radionuclides in Rancho Seco liquid effluent was examined for dose contributions of each radionuclide. It was determined that Cs-134 and Cs-137 contributed nearly 98% of the dose due to gamma emitters to the total body and to specific organs other than the thyroid. I-131 was found to contribute essentially 100 % of the dose to thyroid. Tritium contributes variable fractions of the total dose, but is considered separately due to the distinct analysis method.

2. Dose (Cont.)

The first set of LLDs utilizes four radionuclides as indicators for all other nuclides. Three basic cases are addressed:

1. All gamma emitters other than iodines (Cs-134 & Cs-137)
2. Iodines (I-131)
3. Tritium

This set of LLDs is established at a concentration equivalent of about 50% of the values in Technical Specification 3.17.2, except for Tritium, which is at a concentration equivalent of less than 1% (assuming an estimated annual plant effluent out flow of 20 million gallons and an average dilution flow rate of 8,500 gpm). A comprehensive system which includes administrative limits and a dose tracking program will be used to assess the cumulative offsite calculated dose with respect to the values in Technical Specification 3.17.2 prior to each release.

The second set of LLDs applies to monthly composite samples of the liquid effluent. These LLDs are established at values which represent 10% or less of the concentration equivalent of Technical Specification 3.17.2 (assuming an estimated annual plant effluent out flow of 20 million gallons and an average dilution flow rate of 8,500 gpm). The dose tracking program contains methods for updating the cumulative dose based on results of the composite sample analyses. The mix of radionuclides in the liquid effluent will also be evaluated at semiannual intervals. (Refer to Attachment 3, Bases for Lower Limit of Detection Values for Rancho Seco Liquid Effluents).

Basis For No Significant Hazards Determination

The proposed changes to the Technical Specifications regarding doses due to radioactive material in liquid effluents do not involve a significant hazards consideration because operation of Rancho Seco in accordance with these changes would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. These changes do not significantly alter the safety/accident analysis in the Updated Safety Analysis Report (USAR). Dose and dose commitment to the Maximum Hypothetical Individual due to radioactive materials in liquid effluents are maintained to within the ALARA dose guidelines of 10 CFR 50, Appendix I. Therefore, these changes do not significantly increase the probability or consequences of an accident;

2. Dose (Cont.)

(2) create the possibility of a new or different kind of accident from any previously analyzed. These changes reflect clarification in the Technical Specifications and bases for offsite dose commitment due to plant liquid effluents. Compliance with the ALARA dose guidelines of 10 CFR 50, Appendix I is maintained. Therefore, these changes do not create the possibility of a new or different kind of accident;

(3) involve a significant reduction in a margin of safety. These changes provide reasonable assurance of continued compliance with the ALARA dose guidelines of 10 CFR 50, Appendix I. The dose tracking system is required to be operated and maintained such that operating parameters can be adjusted in accordance with the methodology in the Offsite Dose Calculation Manual (ODCM). They do not allow the calculated dose values to the Maximum Hypothetical Individual, when projected to the end of the quarter and/or year, to exceed the ALARA dose guidelines of 10 CFR 50, Appendix I. The use and definition of the Maximum Hypothetical Individual is pursuant to 10 CFR 50, Appendix I dose methodology and is more conservative than a real dose to a member of the public. Therefore, these changes do not involve a significant reduction in a margin of safety.

3. Radiological Environmental Monitoring Program

The Rancho Seco Radiological Environmental Monitoring Program provides for the collection and analysis of specified numbers of samples of surface water, runoff water, shoreline mud and silt, milk, fish, and several classes of harvested food at specified frequencies (Table 3.22-1). This program is based on the guidance of the model program in NUREG-0472, Rev. 2, and NUREG-0472, Rev. 5 (draft). The sampling locations are described in the REMP Manual. Table 3.22-2 provides reporting levels for radionuclide concentrations in the environmental samples in order to appropriately identify when concentrations of radioactive materials and levels of radiation may be higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways.

The changes represent conformance with the Standard RETS in NUREG-0472, Rev. 2, and NUREG-0472, Rev. 5 (draft). In addition, the radiological environmental monitoring program (REMP) will account for all potential land, water usage, and food radiological exposure pathways that exist downstream from Rancho Seco. The models (Table 3.22-1) will take account for long-term buildup of concentrations of radionuclides in bottom sediment doses due to ingesting aquatic food and direct radiation from long-term buildup of radionuclides on land irrigated

3. Radiological Environmental Monitoring Program (Cont.)

with contaminated water. These models are site-specific and their use is encouraged by RG 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50 Appendix I," is the basis for achieving compliance to 10 CFR 50, Appendix I. The Rancho Seco ODCM follows the guidance provided in Reg. Guide 1.109 and is now separate from the REMP.

Basis For No Significant Hazards Determination

The proposed change to the radiological environmental monitoring Technical Specifications does not involve a significant hazards consideration because operation of Rancho Seco in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. These specifications provide for the measurement of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposure of individuals resulting from station operation. This supplements the radiological effluent monitoring specifications by verifying that measurable concentrations of radioactive materials and levels of radiation are not higher than expected for all potential exposure pathways. This change does not alter the safety/accident analysis of Chapters 11 (11.1.7) and 14, of the USAR and therefore, does not significantly increase the probability or consequences of an accident;

(2) create the possibility of a new or different kind of accident from any previously analyzed. The affected specifications concern environmental monitoring and are consistent with the guidance of Standard RETS in NUREG-0472. Therefore, this change does not create the possibility of a new or different kind of accident;

(3) involve a significant reduction in a margin of safety. The text changes represent clarification and programmatic changes to the Radiological Environmental Monitoring Program. The Radiological Environmental Monitoring Program (REMP) accounts for all significant land, water usage and food radiological exposure pathways that exist downstream from Rancho Seco. Additional sampling points, collection frequencies and reporting level requirements have been added to these specifications along with improvements to the annual land use census. Therefore, this change does not involve a significant reduction in a margin of safety.

4. Land Use Census

The Rancho Seco REMP, utilizing the guidance of NUREG-0472, provides for an annual land use census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modification to the monitoring program are made if required by the results of the census. The changes here will include an addition of liquid pathway surveillance so that existing environmental and societal uses of land surrounding Rancho Seco can be kept current. Identification of gardens in the summer, rather than the middle of winter, will be included in the census to assure a more realistic sampling of gardens. In addition, liquid and gaseous pathways are identified and reportable as land use census dose results which will be included in the Annual Radiological Environmental Operating Report.

Basis For No Significant Hazards Determination

The proposed change to the Technical Specification regarding the Rancho Seco site land use census does not involve a significant hazards consideration because operation of Rancho Seco in accordance with this change would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. This change provides clarification and definition of site boundaries and does not significantly alter the safety/accident analysis in the Updated Safety Analysis Report. This is consistent with Standard Tech Specs NUREG-0472;

(2) create the possibility of a new or different kind of accident from any previously analyzed. This change provides an improved definition of existing geographical and site boundaries and does not create the possibility of a new or different kind of accident;

(3) involve a significant reduction in a margin of safety. This change provides clarification and definition of existing plant boundaries and does not involve a significant reduction in a margin of safety.

5. Reporting, Procedures And Audits

The changes to the Tech Specs 6.5.1.6, 6.5.2.8, 6.8, 6.9.2, and 6.9.5 relates to reporting, procedures and audit requirements as related to radiological effluent monitoring. These changes represent a culmination of format requirements to meet the guidelines of Standard RETS in NUREG-0472.

5. Reporting, Procedures And Audits (Cont.)

Basis For No Significant Hazards Determination

The proposed changes to the Technical Specifications regarding the review, audit, procedures and reporting do not involve a significant hazards consideration because operation of Rancho Seco in accordance with these changes would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. These changes incorporate administrative improvements and do not alter plant design or safety/accident analysis as described in the Updated Safety Analysis Report (USAR). Therefore, these proposed changes do not significantly increase the probability of an accident;

(2) create the possibility of a new or different kind of accident from previously analyzed. These proposed changes increase the scope of the administrative Technical Specifications supporting the control and the monitoring of radioactive materials in plant liquid effluents per Standard RETS in NUREG-0472 and does not create the possibility of a new or different accident;

(3) involve a significant reduction in a margin of safety. These administrative changes are conservative and increase the scope of the Technical Specifications for the review, audit, procedures and reporting of plant liquid effluents related activities. Therefore, these proposed changes do not constitute a significant reduction in a margin of safety.

6. Instrumentation

Radioactive gaseous and liquid effluent monitoring instrumentation monitors and controls the release of radioactive materials in plant effluents. The alarm/trip setpoints for these instruments are calculated in accordance with the methodology contained in the Offsite Dose Calculation Manual (ODCM) to ensure that the limits of 10 CFR 20.106 are maintained. Changes are made to the radioactive and liquid effluent monitoring instrumentation Technical Specifications (RSTS 3.15/4.19 and 3.16/4.20, respectively) to provide clarification, editorial improvements, to establish where applicable conformance with the guidance in NUREG-0472, Rev. 5 (draft) and NUREG-0472, Rev. 2, and to reflect current plant design and operation.

6. Instrumentation (Cont.)

The existing RSTS Table 3.15-1/4.19-1 specify operability and surveillance requirements for the Regenerant Holdup Tank (RHUT) Discharge Line Monitor. This monitor functions as the plant liquid effluent monitor, providing automatic termination of plant liquid releases to ensure compliance with the limits of 10 CFR 20.106. This monitor is being replaced by a monitor downstream of the retention basins (Retention Basin Effluent Discharge Monitor) to more closely conform with the standard RETS. Reasonable operability and surveillance requirements are established for the RHUT total flow monitor to ensure that the total volume of water released from the A & B RHUT (effluent accountability point) to the Retention Basin is known for the determination of offsite dose. Additional changes to the bases of RSTS 3.15 reflect current plant design and operational practices regarding the processing and release of primary and secondary waste water.

Radioactive gaseous effluent monitoring instrumentation surveillance frequencies for the instrument channel calibration in Table 4.20-1 are decreased from monthly to refueling pursuant to the guidance contained in NUREG-0472, Rev. 5 (draft). Notations are added to ensure that the channel test for the Auxiliary Building Stack noble gas activity monitor adequately demonstrates functionality. Allowances is provided for not performing channel testing of the Reactor Building Purge Vent and the Auxiliary Building Stack System effluent flow rate devices when conditions pose a personnel safety hazards. These monitors are located in near proximity to the release manifold of the Main Steam Safety Valves and it is prudent to limit personnel access to this area during power operation.

Basis For No Significant Hazards Determination

The proposed changes to the Technical Specifications regarding the radioactive gaseous and liquid effluent monitoring instrumentation do not involve a significant hazards consideration because operation of Rancho Seco in accordance with these changes would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes provide clarification of alarm/trip setpoint, to ensure that the limits of 10 CFR 20.106 are maintained. The relocation of the liquid effluent monitor allows for more comprehensive monitoring of potential radioactive effluent streams in addition to the Regenerant Holdup tanks. Therefore, these changes do not significantly increase the probability or consequence of an accident previously evaluated;

6. Instrumentation (Cont.)

(2) create the possibility of a new or different kind of accident from any previously analyzed. The change in location of the liquid effluent monitor allows a more comprehensive monitoring and automatic termination of potential radioactive liquid effluent streams in addition to the RHUT discharge flow path. The clarifications to the alarm/trip setpoints ensures that the limit of 10 CFR 20.106 are maintained. There are no changes to system functions and therefore these changes do not create the possibility of a new or different kind of accident.

(3) involve a significant reduction in a margin of safety. These changes ensure that the limit of 10 CFR 20.106 are maintained. Allowances are made for instrument channel test of the Reactor Building Purge Vent and Auxiliary Building Stack System effluent flow rate devices in consideration of personnel safety during normal operation. This allowance doe not significantly reduce the reliability of the flow rate devices. Therefore, these revisions do not involve a significant reduction in a margin of safety.

Safety Analysis For Administrative And Editorial Changes (Item 48):

The changes listed in Tech Spec Item No. 48 are administrative and editorial changes. They are made to improve the overall Technical Specification editorial consistency and format, clarify requirements and correct errors. Changes to Technical Specification Section 6, Administrative Controls, are made to adopt the format with Standard Tech Specs NUREG-0472 and current industry standards.

Basis For No Significant Hazards Determination:

The proposed change does not involve a significant hazards consideration because operation of Rancho Seco in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. This enhanced clarity should decrease the potential for unacceptable consequences or accidents. These are editorial and administrative changes which do not increase the probability or consequence of an accident.

(2) create the possibility of a new or different kind of accident from any previously evaluated. A new or different kind of accident will not be created due to these editorial and administrative changes. These administrative changes do not create the possibility of a new or different kind of accident because of enhanced clarity and document consistency per Standard Tech Specs NUREG-0472.

(3) involve a significant reduction in a margin of safety. These editorial and administrative changes ensure that the Technical Specifications clearly address proper procedural and monitoring control relating to radiological effluent releases and will preserve the margin of safety. Therefore, the administrative changes will not reduce the margin of safety.

ATTACHMENT 2

Administrative and Editorial Changes to Rancho Seco Technical Specifications per Proposed Amendment No. 155

RESPONSIBILITIES (Continued)

h. Performance of special reviews and investigations and reports
138 thereon as requested by the AGM, Nuclear Power Production.

138 i. Review of the Plant Security Plan and changes thereto.

j. Review of the Emergency Plan and changes thereto.

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

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

155 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:

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

b. Render determinations in writing with regard to whether or not
155 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
138 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 The Plant Review Committee shall maintain written minutes of each
138 meeting and copies shall be provided to the AGM, Nuclear Power
Production and the Chairman of the Management Safety Review
Committee.

138→← 6.5.4 (Continued)

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
138→← 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
138→← the MSRC, Deputy General Manager, Nuclear or the General Manager.

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.

155→ m. The PROCESS CONTROL PROGRAM and implementing procedures for
← processing and packaging of radioactive wastes from liquid systems at least once per 24 months.

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

138→ Audit reports of reviews encompassed by Section 6.5.4 shall be forwarded to the General Manager, MSRC Chairman, and to the management positions responsible for the areas reviewed within
← 30 days after completion.

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.

138→ 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
138→ 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.
155→ 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.

138→ 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.

6.8 PROCEDURES (Continued)

6.8.3 Temporary changes to procedure 6.8.1 above may be made provided:

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

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

c. The change is documented, reviewed by the PRC and approved by the Plant Superintendent within seven (7) days of implementation.

6.9 REPORTING REQUIREMENTS

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

Startup Report

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitt following (1) Receipt of an operating license; (2) amendment of the license involving a planned increase in power level; (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier; and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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

155 6.9.2 Radiological Reports

6.9.2.1 Annual Radiological Reports

Annual reports covering the activities of the unit, as described
155 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

155 6.9.2.2.1 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar
155 year shall be submitted prior to May 1 of each year.

155 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.

155

6.9.2.2.2 (Continued)

155 The Annual Radiological Environmental Operating Reports shall
138 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
155 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>

155 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.

155

155 6.9.2.3.1 The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and
155 gaseous effluents and solid waste released from the unit with data summarized on a quarterly basis.

155 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)

155→ 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
155→ activities outside the site boundary (Figures 5.1-3 and 5.1-4) during the report period.

155→ 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
155→ 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 Radwaste 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)

155→ 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 Office of Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the 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 concentrations of the limits specified in Appendix B, Table II 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

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

A. A one-time only, "Narrative Summary of Operating Experience" will be submitted to cover the transition period (calendar year 1977).

B. A Reactor Building Structural integrity report shall be submitted within ninety (90) days of completion of each of the following tests covered by Technical Specification 4.4.2 (the integrated leak rate test is covered in Technical Specification 4.4.1.1).

1. Annual Inspection
2. Tendon Stress Surveillance
3. End Anchorage Concrete Surveillance
4. Liner Plate Surveillance

	C.	Inservice Inspection Program	
	D.	Inoperable Accident Monitoring Instrumentation	30 days (3.5.5)
	E.	Status of Inoperable Fire Protection Equipment	
	F.	Inoperable Emergency Control Room/TSC Ventilation Room Filter System	
	G.	Radioactive Liquid Effluent Dose	30 days (3.17.2)
	H.	Noble Gas Limits	30 days (3.18.2)
	I.	Radioiodine and Particulates	30 days (3.18.3)
155→	J.	Gaseous and Liquid Radwaste Treatment	30 days (3.18.4 and 3.17.4)
←	K.	Radiological Environmental Monitoring Program	30 days (3.22)
155→←	L.	Deleted	
155→←	M.	Solid Radioactive Wastes	30 days (3.21)
	N.	Fuel Cycle Dose	30 days (3.25)
155→←	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.

155→ 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

155→ 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;

   115→ 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.

155→← 6.16 <u>OFFSITE DOSE CALCULATION AND RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUALS</u>

6.16.1 <u>Function</u>

155→ 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
155→ 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
155→← 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)
155→← are those systems described in the facility Updated 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
138→ solid) shall be made by the following method. For the purpose of this
← specification, "major changes' is defined in Specification 6.17.3.

138→ Licensee-initiated changes:

1. The Commission shall be informed of all changes by the inclusion of a suitable discussion of each change in the Semiannual Radioactive Release Report 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;

155→ 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;

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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APPENDIX B

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<u>ATTACHMENT 3</u>

Bases for Lower Limit of Detection Values for
Rancho Seco Liquid Effluents

SACRAMENTO MUNICIPAL UTILITY DISTRICT

OFFICE MEMORANDUM

TO: Files

DATE: December 23, 1987
EP 87-2059

Prepared by: Peter Murphy
Peter Murphy

Reviewed by:
Bill Wadman

Approved by:
Harvey Story

SUBJECT: BASES FOR LLDS

OBJECTIVES
This memorandum documents the bases for the LLDs (Lower Limits of Detection) issued in Proposed Amendment 155 (revision 2) to the Rancho Seco Technical Specifications. It details the methods, assumptions and calculations used to determine a set of LLDs for radiological liquid effluents consistent with the guidance promulgated in 10 CFR 50 Appendix I. These numerical ALARA objectives restrict the dose from liquid effluent to any member of the public to 3 mrem whole body and 10 mrem to any organ.

DESIGN STRATEGY
(1) Determine the relative abundance of radioisotopes found in the activity accounting vessels, A and B Regenerative Holdup Tanks (RHUTs) (2) Calculate the dose resulting from a "unit" release of the isotope mix (3) Backcalculate to find the mix activity required to reach 50 percent of the Appendix I guidance or the "target dose" (4) determine the average concentration of the RHUT contents needed to make the target dose by dividing the activities calculated above by the expected volume of radiological effluent and (5) set the LLDs equal to those concentration based on their relative abundance in the mix. All subcalculations mentioned in this paper refer to the subparts of Calculation EP-LLD (Reference 1).

SOURCE TERM
For the purposes of evaluation, separate the isotopes in the effluent into four groups: tritium, iodine-131, major isotopes, and all other isotopes. Since the amount of tritium released annually is roughly four orders of magnitude larger than the anticipated amounts of the rest of the isotopes, including it in the mix would unnecessarily diminish the resolution of the relative mix fractions.

Previous studies (References 2, 4 and 5) on this issue have shown that iodine-131 provides over 99 percent of the thyroid dose and only a vanishingly small part of the adult whole body dose. Since the child thyroid will be the most restrictive case, Subcalculation 3 determines an LLD for that isotope on that bases. The adult whole body dose is the limiting case for all other isotopes (see the computer output in Subcalculation 5).

Subcalculation 4 contains a graph of the amounts of tritium released annually from Rancho Seco in liquid effluent (Reference 6) in the years 1980-85. The activity shows a very weak correlation to plant availability. The extrema may be ignored as unrepresentative because the minimum in 1980 occurred before the plant experienced chronic, low level steam generator tube leakage, and the maximum happened in a high leakage year prior to plant operation with disposable Condensate Polisher Resins. Averaging the remaining years yields an average value of 75 curies of tritium per annum.

Reference 3 indicates that six isotopes, namely Cs-137, Cs-134, Co-58, Co-60, Mn-54, and I-131, comprise more than 95 percent of the activity (excluding tritium) in Rancho Seco liquid effluents. Moreover, these isotopes deliver approximately 99.7 percent of the associated whole body dose. As a matter of simplicity, specific LLDs will be calculated for tritium, and these six isotopes. The rest will be included in an "other" category. Of the nonmajor isotopes, Zn-65 produces the largest offsite dose per unit activity (Reference 7); therefore, the LLD for those isotopes were set by assuming they had the same dose potential as Zn-65.

The tables in Subcalculation 2 show the amounts of activity released offsite in liquid effluents based on data from semi-annual and annual effluent reports (Reference 6). The graphs also indicates the relative fraction each isotope contributes to the total activity. Data from 1986 is included for reference and will not be used in the analysis since the plant did not operate during that year. Inspection of the data reveal that in the years 1981-82, radiocobalt dominated the mix with those two isotopes (Co-58 and-60) making up roughly 62 and 67 percents of the total activity respectively. However, in 1983-85 the radiocesiums constituted the majority of the mix reaching 66, 72, and 100 percents of the total, respectively.

The LLD for an isotope increases with its relative abundance in the mix. Therefore, mixes with an even distribution will result in lower LLDs. Since the plant has been in cold shutdown for two years, we cannot predict whether fission products or activation products will dominate the effluent mix. For conservatism, we develop the historical mix as the straight average of the five operational years prior to shutdown (see Subcalculation 2). This yields conservative results because it reduces the relative proportion of the cesiums and makes their LLDs lower than if the most recent mix were used.

DOSE PROJECTION METHOD

Dose calculations for the release of liquid effluents were accomplished with the aid of LADTAP computer code, issued by the Nuclear Regulatory Commission, which implements the methods detailed in RegGuide 1.109 (Reference 10). The default parameters contained in the code were replaced by more conservative values from the Rancho Seco area land use census. For further conservatism, the drier summer conditions were assumed.

The plant operations procedures for the Waste Water Disposal system specify in the system limits and precautions (section A.29.3) that the minimum dilution flow offsite will be 5000 gpm or 11.14 cubic feet per second. However, Rancho Seco has made an internal commitment to maintain a 8500 gpm dilution flow after plant startup in early 1988, and this value was used in the calculations.

Note that the "total dose" indicated on the summary sheet does not include all pathways. The dose contribution from the vegetation, milk, and meat pathways must be added separately by hand.

ANNUAL VOLUME OF EFFLUENT

Effluent volume data prior to 1985 does not accurately represent the present configuration of the plant in that changeover to disposable polisher resins at the end of 1984 significantly reduced the water (and activity) produced during regeneration. Data after 1985 are not representative since the plant has been in cold shutdown since December of that year.

A strict extrapolation of the 1985 volume based on availability will be overly conservative since only part of the volume varies with time of operation and reactor power. The volume in 1985 for both radioactive and nonradioactive released was 22.9 Mgals when the plant had a 31 percent availability. Given that criticality is scheduled for late January and full power operation will commence in July at the earliest, an upper bound on the availability may be set by taking the average over the last six years of operation (1980-85). Assuming that the 8.5 Mgals of effluent released in 1987 after two years of cold shutdown is essentially constant and independent of the plant availability, the remainder may be scaled according to the anticipated maximum availability for 1988. This yields a tentative estimate of 22.9 Mgals. This can be reduced by including the water diversions of approximately 2.6 Mgals (Reference 8) to the newly constructed C RHUT for a final best estimate of 19.7 Mgals. This value was rounded up for convenience and conservatism to 20 million gallons.

Note that this value does not include the effluent savings possible with the implementation of ECN-1087. The modification, scheduled for early 1988, proposes to redirect four sources of nonradioactive water from the Condensate Pit Sump (lined to the RHUTs for discharge to the retention basins and release offsite) to the condenser hotwell. At 45 percent availability this translates to a reduction of 8.3 million gallons per year.

IODINE
The RegGuide 1.109 dose equation may be simplified considerably by lumping all of the dispersion, uptake, and dose factors for an isotope i and an organ j into one constant, commonly referred to as A_{ij}. This value may be determined by a computer by entering a unit activity for source term with a unit dilution flow rate into LADTAP. In this case the resulting dose is identical to A_{ij}. Once this value is obtained, the dose for other sources and dilution flows may be calculated with the same accuracy as LADTAP with a simple three factor formula.

Subcalculation 3 determines the Aij value for the isotope I-131 and the receptor organ of thyroid. The activity required to deliver the target dose, in this case 50 percent of 10 mrem, was backcalculated as 4.92 mCi from the three factor formula. The corresponding LLD for I-131 was then 6E-8 uCi/ml which is greater than the value derived in Subcalculation 5. This value was used for I-131 because it contributes only six one hundredths of 1 percent to the adult whole body dose.

TRITIUM
In Subcalculation 4 the estimated amount of tritium was entered in a LADTAP computer run. The actual total dose calculated for this source term was 0.63 mrem. The LLD for tritium was then located at 9E-4 uCi/ml.

Since the annual tritium release is considered constant, the dose contribution has to be subtracted from the target dose. The revised target is then calculated to be 0.87 mrem.

MAJOR ISOTOPES
In Subcalculation 5 the historical mix of major isotopes were specified by their relative abundance to Cs-137. This source term was entered into LADTAP. The resulting whole body dose was 0.959 mrem. Since the dose is linearly proportional to activity, this value could be scaled to reach the revised target dose, and the LLD was calculated as described above.

RESULTS
The results presented in Subcalculation 5 put the LLD for the predominant isotope (Cs-137) at 2.0E-8 uCi/ml and the LLDs other major isotopes decrease from that value to a minimum LLD of 2E-9 uCi/ml for the "other" isotope category. The decline in value for less significant isotopes is necessary to preclude unseen contributions to the offsite dose. For example, if a release showing only Cs-137 at the LLD level were released, the rest of the isotopes may be present in proportion to the historic ratio. The dose from these undetected isotopes could result in a dose more than double that projected for the release.

CONCLUSIONS
The current radiation detection equipment onsite (18% GeLi) can reach an LLD of 1.5E-8 uCi/ml with a 4000 second count time. However, neither this apparatus nor the newly acquired 30% machine can reach the LLDs required. These level are beyond the capability of many well equipped environmental laboratories.

The solution to this problem is to define one LLD (the one for Cs-137) for all isotopes, except tritium. Then assume in the prerelease calculation that all the other isotopes are present in their mix ratios if only Cs-137 were detected in the sample. This would account for their dose contributions in the pre-release calculation without requiring unrealistic detection procedures. Should no Cs-137 be detected, the other isotopes would not be considered. Also note that if an isotope is detected below its LLD value, it will be reported and included in all pre- and post-release dose calcualtions.

Due to the sensitivity of the dose on the isotopic composition of the release, the mix would have to be updated on a monthly basis from the composite sample results analyzed by the environmental laboratory. The LLDs for that laboratory would ordinarily be set with a target dose of 10 percent of the Appendix I guidance; however, one fifth the values listed in the final table of Subcalculation 5 are not practically achievable. As an alternative, the LLDs for the principal gamma emitting nuclides were set at the value of Cs-137 required to make 10% of Appendix I as part of mix. This is reasonable in light of the fact that the whole variety of nuclides have been seen in the composite analysis with much higher required LLDs and that the two cesiums contribute more than 97 percent of the dose.

REFERENCES

(1) Calculation EP-LLD prepared by Peter Murphy, December, 1987.

(2) Development of Target Dose Values for Appendix I Compliance, by R. Oesterling et al, March 1987

(3) Bases for Lower Limit of Detection Values for Rancho Seco Liquid Effluents, by R. Oesterling in support of Proposed Amendment 155 (revision 0) to the Rancho Seco Technical Specifications

(4) Lower Limits of Detection (LLD) and RHUT Releases, a memorandum by E. Bradley to J. McColligan, April 28, 1986

(5) Draft Lower Limits of Detection Study, a memorandum from E. Bradley to D. Kaplan, R. L. Powers, F. Kellie, M. Braun, and R. Columbo on October 29, 1985

(6) Rancho Seco Annual Environmental Reports as amended (1980-1986)

(7) The Offsite Dose Calculation Manual (ODCM), 1985

(8) Calculation EP-CRHUT prepared by Peter Murphy, December 1987

(9) Radiation Protection Liquid Effluent Data Base Manager, complete until October, 1986

(10) Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purposes of Evaluating Compliance with 10 CFR 50 Appendix I, October, 1977.

ATTACHMENT

(1) Calculation EP-LLD-MAIN

cc: RIC 5E.200

bc: EP Files

SACRAMENTO MUNICIPAL UTILITY DISTRICT

OFFICE MEMORANDUM

TO: Files

DATE: December 23, 1987
EP 87-2060

Prepared by: Peter Murphy
Peter Murphy

Reviewed by: [signature]
Bill Wadman

Approved by: [signature] for
Harvey Story

SUBJECT: CALCULATION EP-LLD-MAIN

This calculation and its subparts determine a set of Lower Limits of Detection (LLDs) for radiological liquid effluents consistent with ALARA guidance of 10 CFR 50 Appendix I. It supports the bases document submitted with Proposed Amendment 155 (revision 2) to the Rancho Seco Technical Specifications.

The subcalculations may be summarized as follows:

(1) An estimation of the annual volume of water passing through the accounting vessels (Regenerative Holdup Tanks A and B). Sub1 scales the 1985 volume, the only historic data characteristic of the anticipated plant configuration in 1988, to allow for possibly increased water generation with greater availability.

(2) This part presents the historical data (1981-1986) from the annual reports regarding the relative abundance of isotopes in the liquid effluents. Sub2 then determines a "historical mix" of isotopes as a simple arithmetic mean.

(3) Given that the child thyroid dose proves to be the more limiting case for establishing an LLD for I-131, Sub3 calculates the required LLD for that design objective. This subcalculation also adds more detail to the dose projection method description contained in the bases.

(4) Sub4 estimates the anticipated amount of tritium to be released in liquid effluents during 1988 by averaging the last six years of operation. Based on that source term, it determines the tritium LLD.

(5) Based on the historical mix calculated in Sub2, this subpart determines the required LLDs for all isotopes excluding tritium. The six major isotopes accounting for 99.7 percent of the dose are considered explicitly while the rest are grouped into an "other" category.

The conclusions drawn from these investigations are contained in the bases document itself. Note that all computer runs associated with each calculation are included with it.

CC: RIC 5E.200

bc: EP Files

PM.mjd

ATTACHMENT 4

PROPOSED AMENDMENT NO. 155

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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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.

155→ 1.13 PROCESS CONTROL PROGRAM

PROCESS CONTROL PROGRAM (PCP) - The PROCESS CONTROL PROGRAM shall contain the sampling, analysis, and formulation determination by which SOLIDIFICATION 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)

155→ 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

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

1.18 DOSE EQUIVALENT I-131

155→ 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 <u>not</u> 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.

155→ 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. ←

155→ 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).

155→

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

155→ 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:

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

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

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

Where

155→← A = Reactor coolant activity ($\mu Ci/gm = mCi/Kgm$)

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 ($8.51 \times 10^{-4}\ sec/m^3$)

$\bar{E}$ = 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
155→← $\mu c/gm$) 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.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operability of the turbine cycle during normal operation and for the removal of decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the required steam relief capacity during normal operation and the capability to remove decay heat from the reactor core.

Specification

152→ 3.4.1 The reactor coolant system shall not be brought or remain above 280F with irradiated fuel in the pressure vessel unless the following conditions are met:

A. Capability to remove decay heat by use of two steam generators as specified in 3.1.1.2.

B. One atmospheric dump valve per steam generator shall be operable.

C. A minimum of 250,000 gallons of water shall be available in the condensate storage tank.

D. Two main steam system safety valves are operable per steam generator.

E. Both auxiliary feedwater trains (i.e., pumps and their flow paths) are operable.

F. Both trains of main feedwater isolation on each main feedwater line are operable.

G. Four independent backup instrument air bottle supply systems for ADVs and MFW, SFW, and AFW valves are operable.

← With less than the above required components operable, be on decay heat cooling within 72 hours.

155 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
155 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).

155 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
155 the ACTION shown in Table 3.15-1.

Bases

155 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.

155 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)

155→ 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		
155→ 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
155→ 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 155→ 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.

155- 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 3.15-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	Instrument	Minimum Number of Channels Operable	Action
	1. Reactor Building Purge Vent		
155	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.
155	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. **
155	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.

155→←

Table 3.16-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	Instrument	Minimum Number of Channels Operable	Action
	2. Auxiliary Building Stack		
155→	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

* 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)		
		155 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 155 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 155 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 155 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. **

* 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
155	3.	Auxiliary Building Grade Level Vent (continued)		
155		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.**
155		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.
155		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.

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

155→

3.17 LIQUID EFFLUENTS

3.17.1 Concentration

155→ 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
155→ 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
155→ 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
155→ 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

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

155 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
155 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
155 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
155 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
155 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
155 following tanks shall be limited to less than or equal to 10 Curies,
excluding tritium and dissolved or entrained noble gases:

155 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
155 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
155 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
155 applies to each tank individually.

155 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.

155- 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 0.60 mrem to the total body or 2.0 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
155→ 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:

155→ 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

155→ 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
155→ 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
155→ 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.

155→ 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
155→← 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
155→ 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. ←

155- 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

155- 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.

155

3.18.3 (continued)

Bases (continued)

155- 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.

155- 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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155 **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.

155 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
155 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

155 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
155 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
155 the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

155 3.18.5 Gas Storage Tanks

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

Applicability At all times

Action

155 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
155 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
155 be sampled daily upon reaching the reactor coolant system limiting activity
value (43/Ē) 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

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

<u>Applicability</u> At all times

<u>Action</u>

155 a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.22-1, prepare and submit to the
155 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
155 hazardous conditions, or seasonal unavailability.)

b. With the level of radioactivity in an environmental sampling medium
155 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
155 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
155 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:

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

155 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.

155 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
155 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
155 which the replacement samples were obtained are added to the Radiological Environmental Monitoring Program as replacement locations, if available.

3.22 (continued)

Bases

155 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
155 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.

155 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			
155 155 155	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 radioactivity at least 24 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.

155 * 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			
155→ 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			
155→	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.

155→←

Table 3.22-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
H-3	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
155 leafy vegetation in each of the 16 meteorological sectors within a distance of five miles.

155 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.

<u>Applicability</u> At all times

<u>Action</u>

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently
155 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
155 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
155 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).

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*Broad leaf vegetation sampling may be performed at the site boundary in the
155 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
155 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.

155 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

155→ 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) over 12 consecutive months.

Applicability At all times

Action

155→ 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, over 12 consecutive months 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

155→ 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)

155 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, 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
155 monitoring samples for radioactive materials shall participate in an
Interlaboratory Comparison Program approved by the Commission.

<u>Applicability</u> At all times

<u>Action</u>

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

<u>Bases</u>

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.
155→ 44. a. Process radiation monitoring system	W	Q	R	
b. Area radiation monitoring system ←	W	M	Q	
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. Auxiliary Feedwater Start Circuit				
a. Phase imbalance/Underpower RCP	S	NA	R	
b. Low Main Feedwater Pressure	NA	M	R	
49. Pressurizer Water Level	M	NA	R	
50. Auxiliary Feedwater Flow Rate	M	NA	R	
51. Reactor Coolant System Subcooling Margin Monitor	M	NA	R	
52 DMOV 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	
56. Safety Valve Position Indicator (Backup Detector) Acoustic	M	NA	R	

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

	Item	Check	Frequency
155	1. Reactor coolant	a. Radio-chemical analysis(1) Ē determination (3)(4)(6) b. Gross activity(1) (3) c. Tritium radioactivity d. Chemistry (Cl and O_2) e. Boron concentration f. Fluoride	M Semiannually 3/week M 3/week 2/week 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) b. Iodine analysis(2)(3)	Weekly Weekly
155	6. Concentrated boric acid tank	Boron concentration(5)	2/week and after each makeup
155	7. Spray additive tank	NaOH concentration (3) each makeup	Q and after
	9. 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.

155 4.19 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

155 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.

155 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
155 ensure that the limits of Specification 3.17.1 are met.

Bases

155 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
155 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				
155→ 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

155→

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

155 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.

155 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
155 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
155 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
155 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				
155→←	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
155→←	d. System Effluent Flow Rate Device	D	NA	R	Q(6)
155→←	e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q
	2. Auxiliary Building Stack				
155→←	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
155→←	d. System Effluent Flow Rate Device	D	NA	R	Q(6)
155→←	e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q

155→←

Table 4.20-1 (Continued)

	Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
155→←	3. Auxiliary Building Grade Level Vent				
155→←	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
155→←	d. System Effluent Flow Rate Device	D	NA	R	Q
155→←	e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q

155→←

Table 4.20-1 (Continued)

TABLE NOTATION

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

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

2. 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
155 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
155 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.

155

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
155 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
155 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
155 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.

155 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)
155→ 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

155→ 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 B, 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

155→ 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

155→ 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. ←

155→ 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.

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.

155→

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)

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

←

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.

155→ 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 AN LYSIS 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

4.21.2 Doses

Dose Calculations

155→ 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.
155→← 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
155→ 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

155→←

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.

155→ 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.

4.21.3 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
155 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.

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*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.

4.22 GASEOUS EFFLUENTS

4.22.1 Dose Rate

Surveillance Requirements

155→←
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
155→← limits in Specification 3.18.1.

155→ 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.

155→ 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
155→ 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
155→ 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
155→ 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
155→ 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)
155	A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters (f)	1×10^{-4}
155	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}
155	C. Auxiliary Building Stack	M(b,c,e) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
				H-3	1×10^{-6}
155	D. Auxiliary Building Grade Level Vent	M(b) Grab Sample	M(b)	Principal Gamma Emitters (f)	1×10^{-4}
				H-3	1×10^{-6}
155	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}
155		Continuous	M Composite Particulate Sample	Gross Alpha(h)	1×10^{-11}
				Sr-89, Sr-90(g)	1×10^{-11}
155		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

155→ 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

155→ 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.

155→ 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.

155

155 4.22.2 Dose-Noble Gases

Dose Calculations

155 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
155 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
155 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
155 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
155 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.

155→ 4.22.3 <u>Dose-Iodine-131, Iodine-133, Tritium, and Radioactive Materials in</u>
← <u>Particulate Form</u>.

<u>Dose Calculations</u>

155→ 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.

<u>Bases</u>

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
155→ 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.
155→← These equations also provide for estimating doses based upon the historical average atmospheric conditions. The release rate specifications for
155→← 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.

155→ 4.22.4 <u>Gaseous Radwaste Treatment</u>

<u>Surveillance Requirement</u>

155→ 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.

<u>Bases</u>

155→ 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
155→ the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

155→← 4.22.5 <u>Gas Storage Tanks</u>

<u>Surveillance Requirements</u>

The quantity of radioactive material contained in each waste gas decay tank
155→← 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.

<u>Bases</u>

Restricting the quantity of radioactivity contained in each waste gas decay tank provides assurance that in the event of an uncontrolled release of the
155→ 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
155→← approaches the limits of Specification 3.18.5.

4.25 SOLID RADIOACTIVE WASTES

Surveillance Requirements

155→ 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.

155→ 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

155→ 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.

4.26 RADIOLOGICAL ENVIRONMENTAL MONITORING

Surveillance Requirements

The radiological environmental monitoring samples shall be collected per Table
155 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

155 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
155 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
155 sample locations have been listed in the REMP MANUAL to retain flexibility for
making changes as needed.

155 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)
155→ 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 4.25-1 (Continued)

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

Table Notation

155→ 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)

$= b$ [illegible]

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:

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

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, 5×10^{-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.

155→ 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.

155 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
155 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 required 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.

155 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

155 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).

155 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
155 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

155→ 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

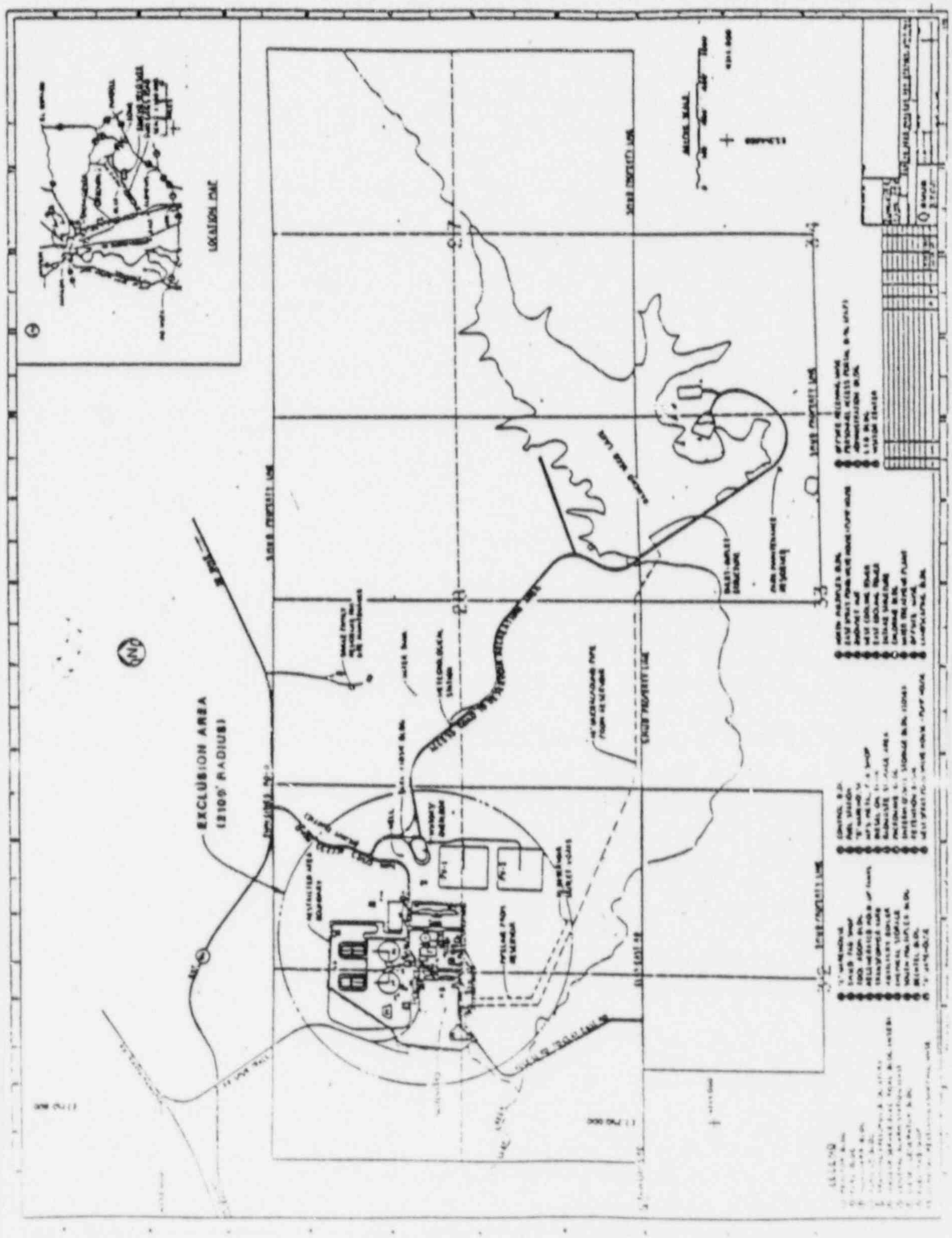

FIGURE 5.1-2

LOW POPULATION ZONE
(5 mile radius)

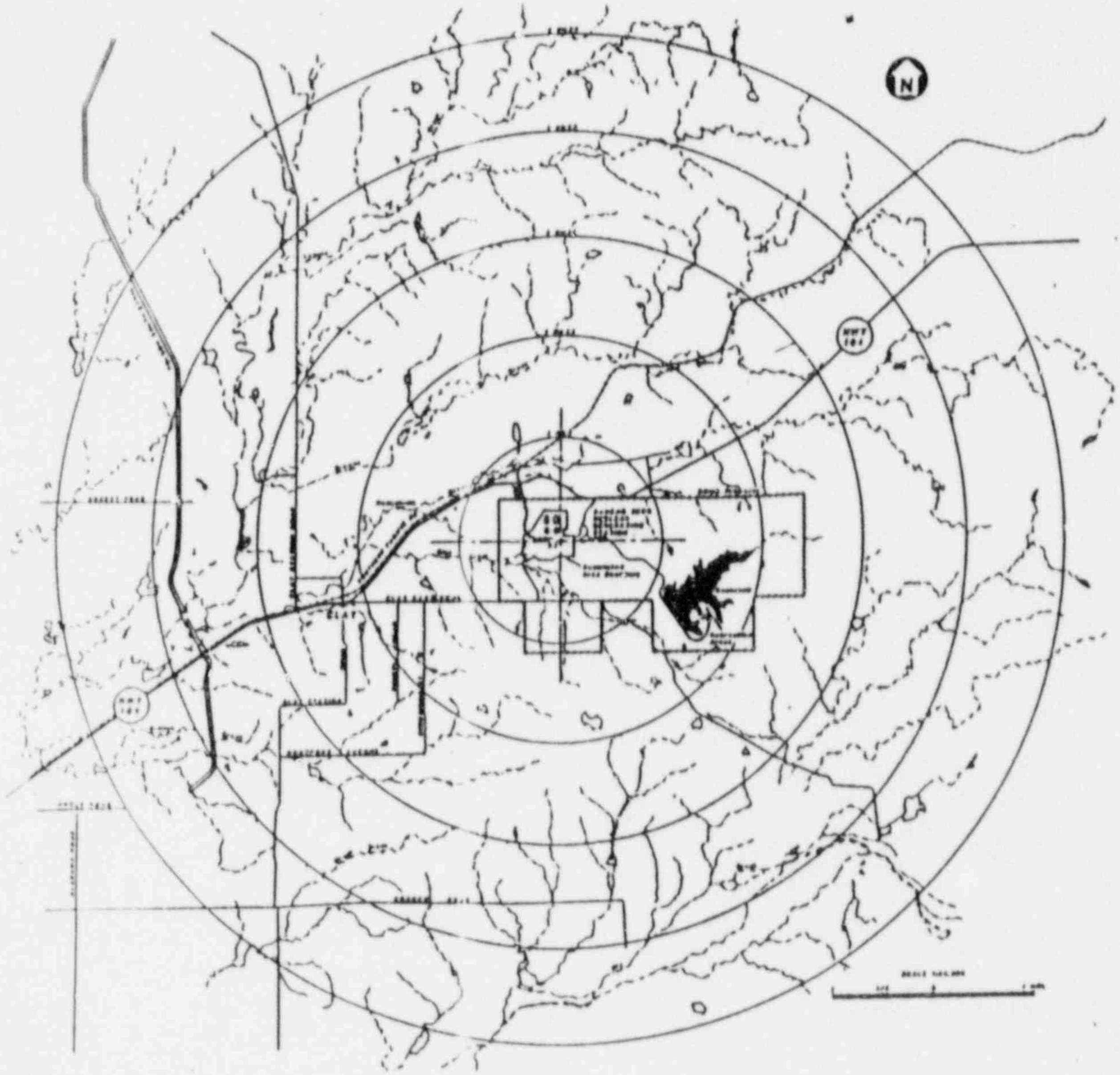

FIGURE 5.1-3

SITE BOUNDARY FOR GASEOUS AND LIQUID EFFLUENTS

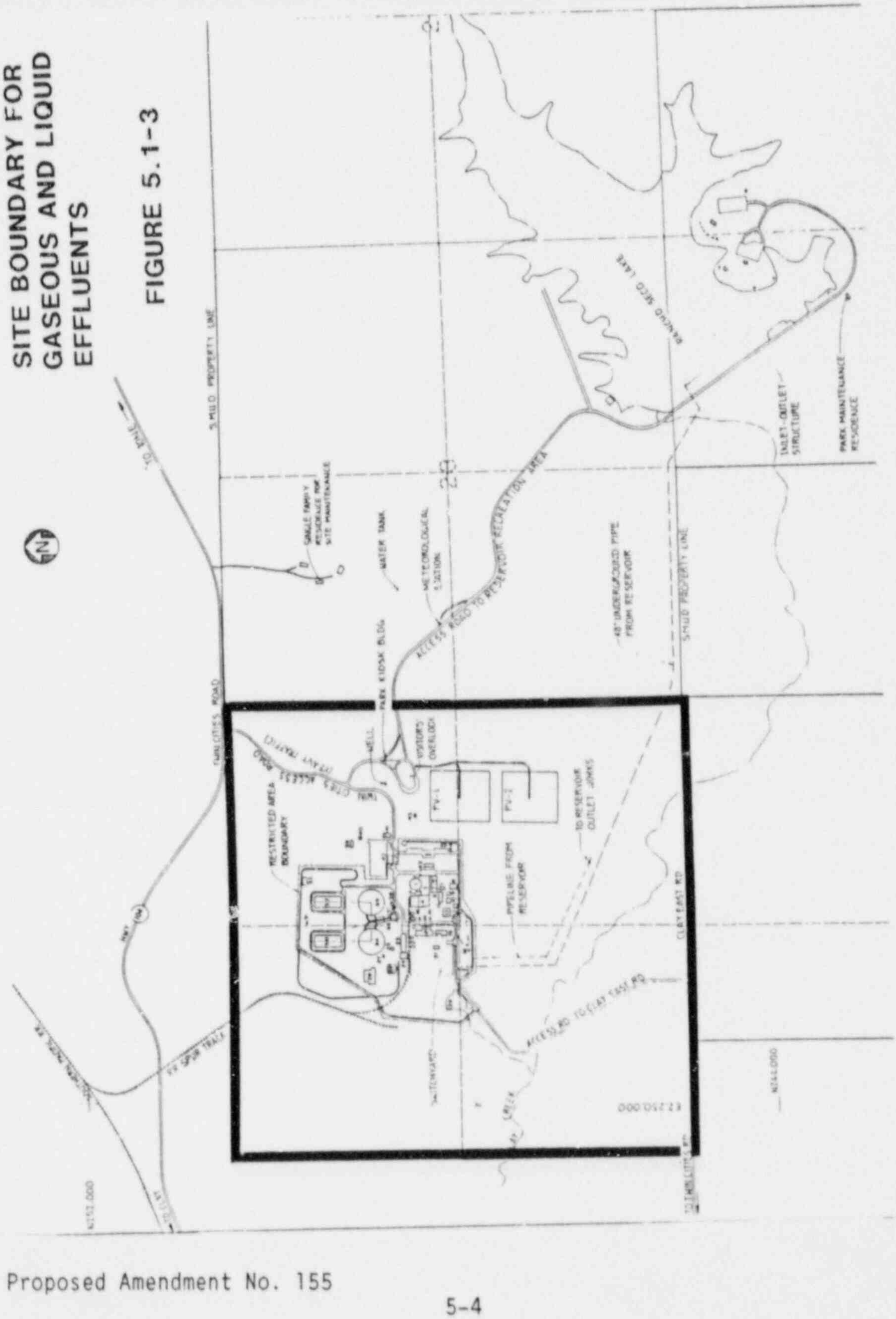

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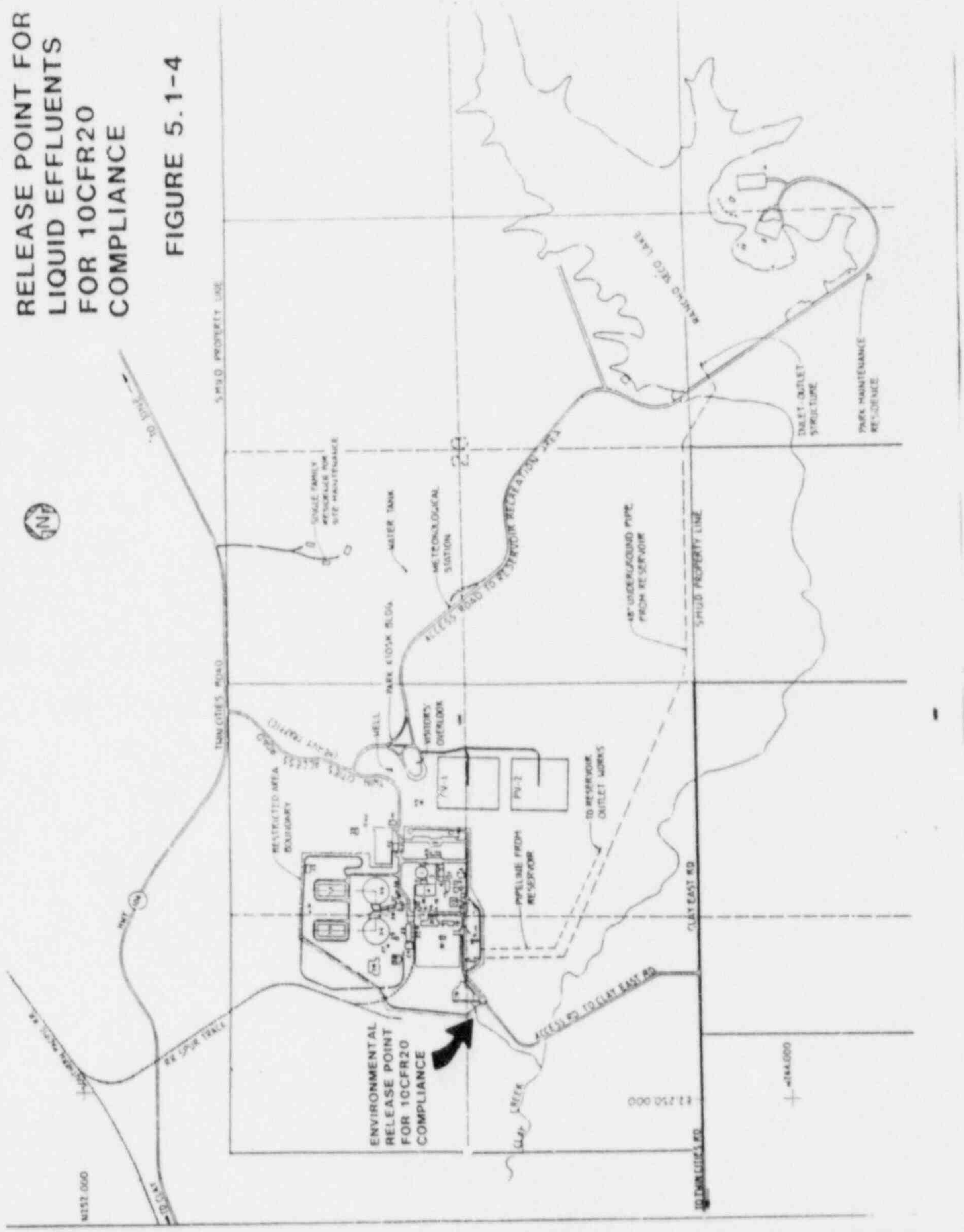
FIGURE 5.1-4

SITE BOUNDARY FOR LIQUID EFFLUENTS FOR 10 CFR 20 COMPLIANCE



RESPONSIBILITIES (Continued)

h. Performance of special reviews and investigations and reports
138 thereon as requested by the AGM, Nuclear Power Production.

138 i. Review of the Plant Security Plan and changes thereto.

j. Review of the Emergency Plan and changes thereto.

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

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

155 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:

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

b. Render determinations in writing with regard to whether or not
155 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
138 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 The Plant Review Committee shall maintain written minutes of each
138 meeting and copies shall be provided to the AGM, Nuclear Power Production and the Chairman of the Management Safety Review Committee.

138→← 6.5.4 (Continued)

.. 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
138→← 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
138→← the MSRC, Deputy General Manager, Nuclear or the General Manager.

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.

155→ m. The PROCESS CONTROL PROGRAM and implementing procedures for
← processing and packaging of radioactive wastes from liquid systems at least once per 24 months.

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

138· Audit reports of reviews encompassed by Section 6.5.4 shall be forwarded to the General Manager, MSRC Chairman, and to the management positions responsible for the areas reviewed within
← 30 days after completion.

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.

138→ 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
138→ 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.
155→ 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.

138→ 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.

6.8 PROCEDURES (Continued)

6.8.3 Temporary changes to procedure 6.8.1 above may be made provided:

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

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

c. The change is documented, reviewed by the PRC and approved by the Plant Superintendent within seven (7) days of implementation.

6.9 REPORTING REQUIREMENTS

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

Startup Report

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitt following (1) Receipt of an operating license; (2) amendment of the license involving a planned increase in power level; (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier; and (4) modifications that may have significantly altered the nuclear, thermal or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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

155 6.9.2 <u>Radiological Reports</u>

6.9.2.1 <u>Annual Radiological Reports</u>

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

6.9.2.1.1 <u>Annual Occupational Radiation Exposure Report</u>

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 <u>Annual Exposure Report</u>

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 <u>Annual Radiological Environmental Operating Report</u>

155 6.9.2.2.1 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar
155 year shall be submitted prior to May 1 of each year.

155 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.

155

6.9.2.2.2 (Continued)

155 The Annual Radiological Environmental Operating Reports shall
138 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
155 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>

155 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.

155

155 6.9.2.3.1 The Semiannual Radioactive Effluent Release Reports shall
include a summary of the quantities of radioactive liquid and
155 gaseous effluents and solid waste released from the unit with
data summarized on a quarterly basis.

155 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)

155 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
155 activities outside the site boundary (Figures 5.1-3 and 5.1-4) during the report period.

155 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
155 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)

155→ 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 Office of Director of Inspection and Enforcement, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the 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 concentrations of the limits specified in Appendix B, Table II 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

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

A. A one-time only, "Narrative Summary of Operating Experience" will be submitted to cover the transition period (calendar year 1977).

B. A Reactor Building Structural integrity report shall be submitted within ninety (90) days of completion of each of the following tests covered by Technical Specification 4.4.2 (the integrated leak rate test is covered in Technical Specification 4.4.1.1).

1. Annual Inspection

2. Tendon Stress Surveillance

3. End Anchorage Concrete Surveillance

4. Liner Plate Surveillance

C. Inservice Inspection Program

D. Inoperable Accident Monitoring Instrumentation 30 days (3.5.5)

E. Status of Inoperable Fire Protection Equipment

F. Inoperable Emergency Control Room/TSC Ventilation Room Filter System

G. Radioactive Liquid Effluent Dose 30 days (3.17.2)

H. Noble Gas Limits 30 days (3.18.2)

I. Radioiodine and Particulates 30 days (3.18.3)

155→ J. Gaseous and Liquid Radwaste Treatment 30 days (3.18.4 and 3.17.4)

← K. Radiological Environmental Monitoring Program 30 days (3.22)

155→← L. Deleted

155→← M. Solid Radioactive Wastes 30 days (3.21)

N. Fuel Cycle Dose 30 days (3.25)

155→← 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.

155→ 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

155 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
    115 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.

155 6.16 OFFSITE DOSE CALCULATION AND RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUALS

6.16.1 Function

155 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
155 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
155 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

155 The radioactive waste treatment system (liquid, gaseous, and solid) are those systems described in the facility Updated 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.

138 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.

138 Licensee-initiated changes:

1. The Commission shall be informed of all changes by the inclusion of a suitable discussion of each change in the Semiannual Radioactive Release Report 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;

155 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;

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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APPENDIX B

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