APPENDIX A

TO

TECHNICAL SPECIFICATIONS

FOR THE

RANCHO SECO UNIT 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT

DOCKET NO. 50-312

Revised to incorporate NUREG-0472

May 1979

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TECHNICAL SPECIFICATIONS

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1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a heat balance considering all heat losses and additions.

1.5.7 Source Check

A source check is the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.6 QUADRANT POWER TILT

Quadrant to average power tilt is expressed in percent as defined by the following equation:

$$100 \left(\frac{\text{Power in any core quadrant}}{\text{Average for all quadrants}} - 1 \right)$$

1.6.1 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core.

1.7 CONTAINMENT INTEGRITY

Containment integrity exists when the following conditions are satisfied:

A. The equipment hatch is closed and sealed and both doors of the personnel hatch and emergency hatch are closed and sealed except in B below.

B. At least one door on each of the personnel hatch and emergency hatch is closed and sealed during refueling operations or personnel passage through these hatches.

C. All non-automatic containment isolation valves and blind flanges are closed as required.

D. All automatic isolation valves are operable or closed in the safety features position.

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E. The containment leakage satisfies Specification 4.4.1 and no known changes have occurred.

1.8 REPORTABLE OCCURRENCE

Defined under Administrative Controls Section 6.9.5.

1.9 TIME PERIODS

May be extended to a maximum of +25% to accomodate operations scheduling. The total maximum combined interval time for any three consecutive tests shall not exceed 3.25 times a single specified surveillance interval.

1.9.1 Shifts (S)

A time period covering at least once per twelve (12) hours.

1.9.2 Daily (D)

A time period spaced to occur at least once per twenty-four (24) hours.

1.9.3 Weekly (W)

A time period spaced to occur at least once per seven (7) days.

1.9.4 Fortnightly (F)

A time period covering two consecutive weeks spaced to occur 26 times a year.

1.9.5 Monthly (M)

A time period spaced to occur at least once per thirty-one (31) days.

1.9.6 Quarterly (Q)

A time period spaced to occur at least once per ninety-two (92) days.

1.9.7 Semi-Annually (S.Y.)

A time period spaced to occur at least once per six (6) months.

1.9.8 Annually (A)

A time period spaced to occur at least once per twelve (12) months.

1.9.9 Biennially (B.Y.)

A time period spaced to occur at least once in two (2) years.

357 015

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 systems is assured.

1.14 SOLIDIFICATION

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

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.

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

The boundary of the SMUD owned property.

357 0..

357 016

3.18 FUEL CYCLE DOSE

The dose or dose commitment to a real individual from all uranium fuel cycle sources is limited to $\leq$25 mrem to the total body or any organ (except the thyroid, which is limited to $\leq$75 mrem) over a period of 12 consecutive months.

Applicability: At all times.

Action:

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.22.1.a, 3.22.1.b, 3.22.2.1, 3.22.2b, 3.22.3a, or 3.22.3b, prepare and submit a Special Report to the Commission and limit the subsequent releases such that the dose or dose commitment to a real individual from all uranium fuel cycle sources is limited to $\leq$25 mrem to the total body or any organ (except thyroid, which is limited to $\leq$75 mrem) over 12 consecutive months. This Special Report shall include an analysis which demonstrates that radiation exposures to all real individuals from all uranium fuel cycle sources (including all effluent pathways and direct radiation) are less than the 40 CFR Part 190 Standard. Otherwise, obtain a variance from the Commission to permit releases which exceeds the 40 CFR Part 190 Standard.

Bases:

This specification is provided to meet the reporting requirements of 40 CFR 190.

357 017

3.19 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.19-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.21 are not exceeded.

Applicability: During radioactive releases via the pathways identified in Table 3.19-1.

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 3.21 are met, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.19-1.

Bases

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. 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 the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

357 018

TABLE 3.19-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Action
1. Gross Radioactivity Monitors Providing Automatic Termination of Release	
a. Regenerant Hold-Up Tank Discharge Line Monitor	With the monitor inoperable effluent releases may be resumed for up to 14 days provided that prior to initiating a release: 1. At least two independent samples are analyzed in accordance with Specification 3.21. 2. A second member of the facility staff will independently verify the release rate calculations and discharge valving.
2. Flow Rate Measurement Devices	
a. Regenerant Hold-up Tank Discharge Line	*
b. Waste Water Flow	With the flow rate measurement device inoperable, effluent releases via this pathway may continue for up to 14 days provided the flow rate is estimated and recorded at least once per four hours during actual releases.
c. Regenerate Hold-up Tank Radiation Monitor Flow	Repair flow monitor to permit continued release.

*Pump curves are utilized to estimate flow.

357 019

3.20 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.20-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.22 are not exceeded.

Applicability: During release via the pathways identified in Table 3.20-1.

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 3.22 are met, declare the channel inoperable.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 3.20-1.

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 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.

357 020

Table 3.20-1

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Action
1. Reactor Building Purge Vent	
a. Noble Gas Activity Monitor	With the monitor inoperable, effluent releases via this pathway may continue for up to 28 days provided grab samples are taken at least once per 8 hours and are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Sampler	With the collection device inoperable, effluent releases via this pathway may continue for up to 28 days provided grab samples are taken at least once per 8 hours 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 for up to 28 days provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours
d. System Effluent Flow Rate Measurement Device	With the flow rate device inoperable, effluent releases may continue for up to 28 days provided the flow rate is estimated and recorded at least once per 4 hours.
e. Sampler Flow Rate Measurement Device	With the flow rate device inoperable, effluent releases via this pathway may continue for up to 28 days provided the flow rate is estimated and recorded at least once per 4 hours.

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TABLE 3.20-1 (continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Action
2. Auxiliary Building Stack	
a. Noble Gas Activity Monitor	With the monitor inoperable, effluent releases via this pathway may continue for up to 28 days provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours
b. Iodine Sampler	With the collection device inoperable effluent releases via this pathway may continue up to 28 days provided grab samples are taken at least once per 8 hours 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 up to 28 days provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
d. System Effluent Flow Rate Measurement Device	With the flow rate device inoperable effluent releases via this pathway may continue up to 28 days provided the flow rate is estimated and recorded at least once per 4 hours.
e. Sampler Flow Rate Measurement Device	With the flow rate device inoperable effluent releases via this pathway may continue up to 28 days provided the flow rate is estimated and recorded at least once per 4 hours.

357 022

TABLE 3.20-1(continued)

RADIOACTIVE GASES EFFLUENT MONITORING INSTRUMENTATION

Instrument	Action
3. Radwaste Service Area Vent	
a. Noble Gas Activity Monitor	With the monitor inoperable, effluent releases via this pathway may continue up to 28 days provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
b. Iodine Sampler	With the collection device inoperable effluent releases via this pathway may continue up to 28 days provided grab samples are taken at least once per 8 hours 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 up to 28 days provided grab samples are taken at least once per 8 hours and these samples are analyzed in accordance with Table 4.22-1 within 24 hours.
d. System Effluent Flow Rate Measurement Device	With the flow device inoperable effluent releases via this pathway may continue up to 28 days provided the flow rate is estimated and recorded at least once per 4 hours.
e. Sampler Flow Rate Measurement Device	With the flow rate device inoperable effluent releases via this pathway may continue up to 28 days provided the flow rate is estimated and recorded at least once per 4 hours.

357 023

3.21 LIQUID EFFLUENTS

3.21.1 Concentration

The co centration of radioactive material released at anytime beyond the site boundary 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 $2x10^{-4}\mu Ci/ml$ total activity.

Applicability: At all times

Action:

With the concentration of radioactive material released from the site to unrestricted areas exceedi g 10 CFR 20.403 restore concentration within the above limits and provide notification to the Commission within 24 hours and a Licensee Event Report within two (2) weeks.

3.21.2 Dose

The dose or dose commitment to an individual from radioactive materials in liquid effluents released beyond the site boundary shall be limited:

a. During any calendar quarter to 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 from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report. This report will identify the cause (s) for exceeding the limit and define the corrective actions to be taken to reduce the releases so that the average dose or dose commitment to an individual from such releases during the current and subsequent three calendar quarters is within 3 mrem to the total body and 10 mrem to any organ.

357 024

3.21 (Continued)

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 10.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioiotope 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.

357 025

RADIOACTIVE EFFLUENTS

3.22 GASEOUS EFFLUENTS

3.22.1 Dose Rate

The dose rate 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 all radioiodines and for all radioactive materials in particulate form and radionuclides other than noble gases 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 to comply with the limit (s) given in Specification 3.22.1 and provide notification to the Commission within 24 hours and a Licensee Event Report within 2 weeks.

Bases

This specification is provided to ensure that the dose rate at anytime at the restricted area boundary (see Figure 3.21-1) from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 beyond the site boundary areas. The annual dose limits are the doses associated with the concentration of 10 CFR Part 20, Appendix B, Table II. 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/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 an infant via the cow-milk-infant pathway to 1500 mrem/yr for the nearest cow to the plant.

357 026

3.22.2 Noble Gases

The air dose 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 any calendar year, to 10 mrad for gamma radiation and 20 mrad for beta radiation.

Applicability: At all times

Action:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission with 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 so that the average dose during the current and subsequent three calendar quarters is within (10) mrad for gamma radiation and (20) mrad for beta radiation.

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 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 Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, " Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision 1, July 1977. The ODCM equations provided for determining the air doses at or beyond the restricted area boundary (see Figure 3.21-1) 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.

357 027

3.22.3 Radioiodines and Particulates

The dose to an individual from radioiodines and particulates with half-lives greater than eight days in gaseous effluents released 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

a. With the calculated dose from the release of radioiodines and particulates 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 and define the corrective actions to be taken to reduce the releases so that the average dose or dose commitment to an individual from such releases during the current and subsequent three calendar quarters is within 15 mrem to any organ.

Bases

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 FCR 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. The ODCM calculational methods approved by 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 I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors", Revision I, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions.

357 028

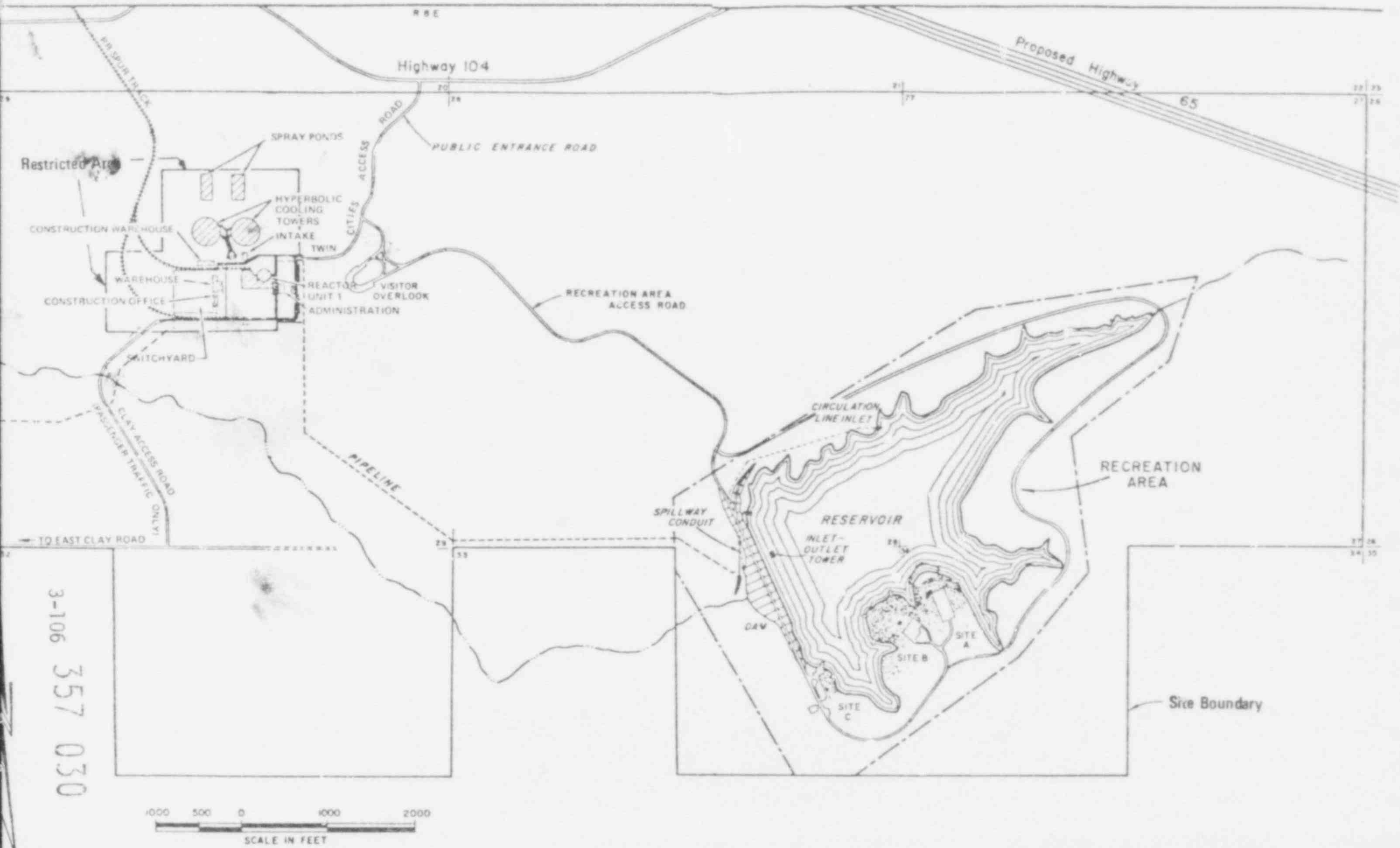
The release rate specifications for radioiodines and particulates are dependent on the existing radionuclide pathways to man, beyond the site boundary. The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of 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.

357 029

POOR ORIGINAL

Limiting Conditions for Operation

FIGURE 3.22-1
GENERAL LAYOUT OF SITE



 357 030

3.23 GASEOUS RADWASTE TREATMENT

The gaseous radwaste treatment and/or the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the sum of the cumulated dose to date for the quarter and the projected doses for the remainder of the quarter would result in doses exceedir 25% of the limits of Specification 3.22.

Applicability: At all times

Action:

a. With gaseous waste being discharged for more than 31 days 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. Identification of equipment of subsystems not OPERABLE and the reason for inoperability.

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

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

Bases

The OPERABILITY of the 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 is reasonably achievable." The specification implements the requirements of 10 CFR Part 50.36 A, 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.

357 031

3.24 GAS STORAGE TANKS

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

Applicability: At all times

Action

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 and provide notification to the Commission within 24 hours and a Licensee Event Report within 2 weeks.

Bases

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 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, and 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 Section 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 limit on coolant activity (Section 3.1.4) 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 coolant limiting activity value ($43/\bar{E}$) to insure 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.

357 032

3.25 SOLID RADIATION WASTES

The solid radwaste systems shall be OPERABLE and used to provide for the SOLIDIFICATION of liquid wastes, for the SOLIDIFICATION and packaging of other radioactive wastes and to ensure the meeting of the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

Applicability At all times.

Action

a. With the requirements of 10 CFR Part 20, 10 CFR Part 71, and the PROCESS CONTROL PROGRAM of specification 6.14 not satisfied, suspend shipments of defective containers of solid radioactive wastes from the site.

b. With the solid radwaste system not OPERABLE for more than 31 days, when required to meet 10 CFR Part 20 and 10 CFR Part 71, to Specification 6.9.2, submit a Special Report which includes the following information:

1. Identification of equipment or subsystems not OPERABLE and the reasons for inoperability.

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

3. A description of alternatives used for SOLIDIFICATION and packaging of wastes.

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

If the solidification system is not required for waste elimination it will be noted in the monthly report.

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 Part 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.

357 033

3.26 RADIOLOGICAL ENVIRONMENTAL MONITORING

The radiological environmental monitoring program shall be conducted as specified in Table 3.26-1.

Applicability: At all times

Action:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.26-1, 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, 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 at one or more of the locations specified in Table 3.26-1 exceeding the limits of Table 3.26-2 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 which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits to be exceeded. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from any of the sample locations required by Table 3.26-1, 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 3.26-1 provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations, if available.

357 034

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 thereby supplements the radiological effluent monitoring program by verifying that the measureable concentrations of radioactive materials a . 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.

357 035

TABLE 3.26-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
A. Radioiodine and Particulates	Locations 1,2,6,8, 10,14,16,17	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 greater than or equal to 24 hours following filter change. Perform gamma isotopic analysis on composite (by location) sample at least once per quarter.
2. DIRECT RADIATION	Locations 1,2,6, 8-14,16,17	At least once per quarter.	Gamma dose. At least once per quarter.

* Sample locations are shown on Figure 3.26-1.

357 036

TABLE 3.26-1 (continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Sample Location*	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	Locations 1,4,8	Grab sample collected monthly.	Gross Beta analysis of each suspended and dissolved fraction. Tritium analysis at least once per quarter.
b. Runoff	Location 1	Grab sample collected fortnightly.	Gross Beta analysis of each suspended and dissolved fraction. Tritium analysis at least once per quarter, plus gamma isotopic analysi on dissolved and suspended fractions.
c. Well	Locations 1,6,11	At least once each quarter.	Gross Beta and Tritium analysis of each sample.
d. Drinking	Location 1	At least once each quarter.	Gross Beta and Tritium analysis of each sample.
e. Mud and Silt	Location 1	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 on Figure 3.26-1.

357 037

Limiting Conditions for Operation

TABLE 3.26-1 (continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Sample Locations*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	Locations 1,3,9,12	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	Location 1 One sample of each of the following species: 1. Red Eared Sunfish 2. Bass	At least semi-annually.	Gross Beta minus K-40 analysis on edible portion of each sample.
c. Food Products	Locations 5,7,10, 11	At time of harvest. One sample of each of the following classes of food products: 1. Leafy vegatables 2. Fleshy vegetables or fruits	Gross Beta minus K-40 analysis on edible portion of each sample.

*Sample locations are shown on Figure 3.22-1.

357 038

TABLE 3.26-2

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	3×10^4				
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			
Cs-137	50	20			
Ba-La-140	2×10^2				
Gross beta	40	2	1×10^2		1×10^2

357 039

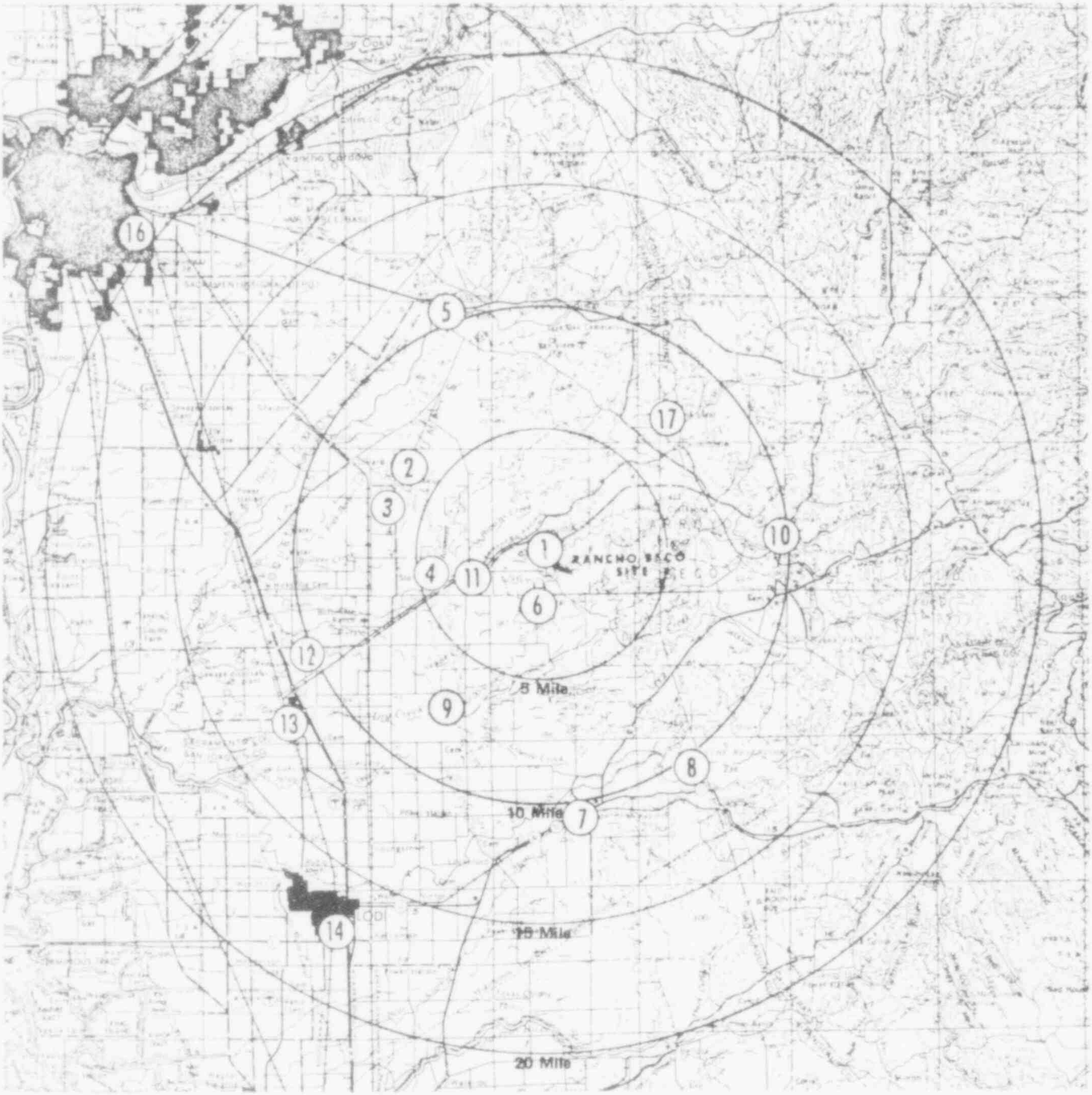


Figure 3.26-1

SAMPLE COLLECTION LOCATION STATIONS

3.27 LAND USE CENSUS

A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles.

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, prepare and submit to the Commission within 30 days a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) greater than at a location from which samples are currently being obtained in accordance with Specification 3.26, prepare and submit to the Commission within 30 days a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days, if possible, The sampling location having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after the growing season of the year in which this land use census was conducted.

*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.

357 041

Bases

This specification is provided to ensure that changes in the use of areas 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/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.

357 042

3.28 EXPLOSIVE GAS MIXTURE

The concentration of oxygen in the waste gas holdup system shall be limited to $\leq$4% by volume.

Applicability

At all times.

Action

With the concentration of oxygen in the waste gas holdup system > 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to $\leq$ 4% within 48 hours.

Bases

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

357 043

4.19 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

Surveillance Requrements

The setpoints 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 channel shall be demonstrated OPERABLE by performance of the INSTRUMENT CHANNEL CHECK, SOURCE CHECK, INSTRUMENT CHANNEL CALIBRATION, AND CHANNEL TEST at the frequencies shown in Table 4.19-1.

Records shall be maintained in the Process Standards of all radioactive liquid effluent monitoring instrumentation alarm/trip setpoints. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.21 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 with NRC approved 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.

357 044

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	R (2)	Q (3)
2. Flow Rate Monitors				
a. Waste Water Flow	D (4)	NA	NA	NA
b. Radiation monitor Flow	D (4)	NA	NA	Q (3b)

Table Notation

(1) During releases via this pathway a monitor 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.

(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.

357 045

4.20 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Surveillance Requirements

The 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. Setpoints and setpoint calculations shall be available for review to ensure that the limits of Specification 3.22 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 CFR Part 50.

357 016

Table 4.20-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE RFQUIREMENTS

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	M	Q (2)	Q (3)
d.	System Effluent Flow Rate Measurement Device	W	NA	BY	A
e.	Sampler Flow Rate Measurement Device	W	NA	BY	A
2.	Auxiliary Building Stack				
a.	Noble Gas Activity Monitor	D (1)	M	R (2)	Q (4)
b.	Iodine Sampler	W	NA	NA	NA
c.	Particulate Sampler	W	M	Q (2)	Q (4)
d.	System Effluent Flow Rate Measurement Device	W	NA	BY	A
e.	Sampler Flow Rate Measurement Device	W	NA	BY	A

357 047

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	R (2)	Q (4)
b. Iodine Sampler	W	NA	NA	NA
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Measurement Device	W	NA	BY	A
e. Sampler Flow Rate Measurement Device	W	NA	BY	A
4. Waste Gas Hold-up System Oxygen Monitor	D	NA	Q (5)	M

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/strip setpoint.

b. Circuit failure.

(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.

357 048

Table 4.20-1 (continued)

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test

(5) The Channel Calibration shall include the use of standard gas samples containing:

a. Nominal zero volume percent oxygen, balance nitrogen.

b. Nominal four volume percent oxygen, balance nitrogen.

357 049

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 3.19-1.

The liquid effluent continuous monitors having provisions for automatic termination of liquid releases, as listed in Table 3.19-1, 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 3.21.

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 3.21.

Post-release analyses of samples from batch relases 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 concentrations at the point of release are limited to the values in Specification 3.21.

Bases

This specification is provided to ensure that the concentration of radioactive materials relased 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 10.106(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 the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

357 050

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)	Each Batch	Each Batch	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	1×10^{-5}
	Each Batch	M Composite(d)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Each Batch	Q Composite(d)	Sr-89, Sr-90	5×10^{-8}

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.

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 at being present at the LLD level. When unusual circumstances result in LLD's higher than specified, the reasons shall be documented in the semi-annual Radioactive Effluent Release Report.

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.

357 051

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 once per month.

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 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 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. NUREG-0133 provides the methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

357 052

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 3.22.

The noble gas effluent continuous monitors, as listed in Table 3.20-1 shall be used to limit offsite doses within the values established in Specification 3.22 when monitor setpoint values are exceeded.

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 beyond the site boundary, due to radioactive materials other than noble gases 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.

Bases

This specification is provided to ensure that the dose rate at any time at the restricted area boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 beyond the site boundary. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix, B Table II. 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 an infant via cow-milk-infant pathway to 1,500 mrem/year for the nearest cow to the plant.

357 053

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

357 054

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. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and XE-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLD's higher than required, the reasons shall be documented in the semi-annual Radioactive Effluent Release Report.

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

d. Tritium grab samples shall be taken at least once per seven days from the ventilation exhaust from the auxiliary building stack during refueling.

e. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

357 055

4.22.2 Noble Gases

Surveillance Requirements

Dose Calculations:

Cumulative dose contributions for the total time period shall be determined in accordance with the Offsite Dose Calculation Manual (ODCM) at least once every month.

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 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 restricted area 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.

357 056

4.22.3 Radioiodines and Particulates

Surveillance Requirements

Dose Calculations:

Cumulative dose contributions for the total time period shall be determined in accordance with ODCM at least once every month.

Bases:

This specification is provided to implement the requirements of Sections IIC, 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 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.

357 057

4.23 GASEOUS RADWASTE TREATMENT

Surveillance Requirements

Doses due to gaseous releases to unrestricted areas shall be projected at least once per month.

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.

357 058

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 3.24 at least once per day 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 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 storage tank activity approaches the limits of Specification 3.24.

357 059

4.25 SOLID RADIOACTIVE WASTES

Surveillance Requirements

The solid radwaste system shall be demonstrated OPERABLE at least once per quarter, when required to meet 10 CFR 20 and 10 CFR 71, or show the capability for SOLIDIFICATION of the waste by meeting one or more of the conditions below:

a. By performance of functional tests of the equipment and components of the solid radwaste system.

b. By operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM.

C. Verification of the existence of a valid contract for SOLIDIFICATION to be performed in accordance with a PROCESS CONTROL PROGRAM.

The PROCESS CONTROL PROGRAM of Specification 6.14 shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of liquid radioactive waste. The test specimens shall be processed in the radiochemical or waste processing laboratory in accordance with procedures of the PROCESS CONTROL PROGRAM:

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 waste until three consecutive test samples demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.14, to assure SOLIDIFICATION of subsequent batches of waste.

357 060

4.25 Solid Radioactive Wastes (Continued)

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).

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 Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 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.

357 061

4.26 RADIOLOGICAL ENVIRONMENTAL MONITORING

Surveillance Requirements

The radiological environmental monitoring samples shall be collected per Table 3.26-1 from the locations shown on Figure 3.26-1 and shall be analyzed to the requirements of Tables 3.26-1 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 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 the present time. Program changes may be initiated based on operational experience.

357 062

Table 4.26-1

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

Analysis	Water (pCi/l)	Airborne Particulate or Gas (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					
^{59}Fe	30					
$^{58,60}Co$	15					
^{65}Zn	30					
^{95}Zr-Nb	15					
^{131}I	1	7×10^{-2}		1		
$^{134,137}Cs$	16 18	1×10^{-2} c				
^{140}Ba-La	15					

357 063

Table 4.26-1 (Continued)

Table Notation

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

357 064

4.27 LAND USE CENSUS

Surveillance Requirements

The land use census shall be conducted at least once per annum by 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 beyond the 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.

357 065

4.28 EXPLOSIVE GAS MIXTURE

Surveillance Requirements

The concentration of oxygen in the waste gas hold-up system shall be determined to be within the limits specified in 3.28 by continuously monitoring the waste gases in the waste gas hold-up tank with the oxygen monitor required OPERABLE according to table 4.20-1.

Bases:

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

357 066

4.29 FUEL CYCLE DOSE

Surveillance Requirements

Cummulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 3.22.1.a, 3.22.1.b, 3.22.2.a, 3.22.2.b, 3.22.3.a, and 3.22.3.b, and in accordance with the Offside Dose Calculation Manual (ODCM).

Reports

Special Reports shall be submitted as required under Specification 3.18.

Bases

This specification is provided to meet the reporting requirements of 40 CFR 190.

357 067

TABLE 6.2-1

SHIFT CREW PERSONNEL AND LICENSE REQUIREMENTS

RANCHO SECO JOB TITLE	REACTOR MODE COLD SHUTDOWN	REACTOR MODE OTHER THAN COLD SHUTDOWN
Shift Supervisor	1 - SL	1 - SL
Sr. Control Room Operator or Control Room Operator	1 - L	2 - L*
Auxiliary Operator or Equipment Attendant	1	1
Equipment Attendant or Power Plant Helper		1
Minimum Total Personnel**	3	5

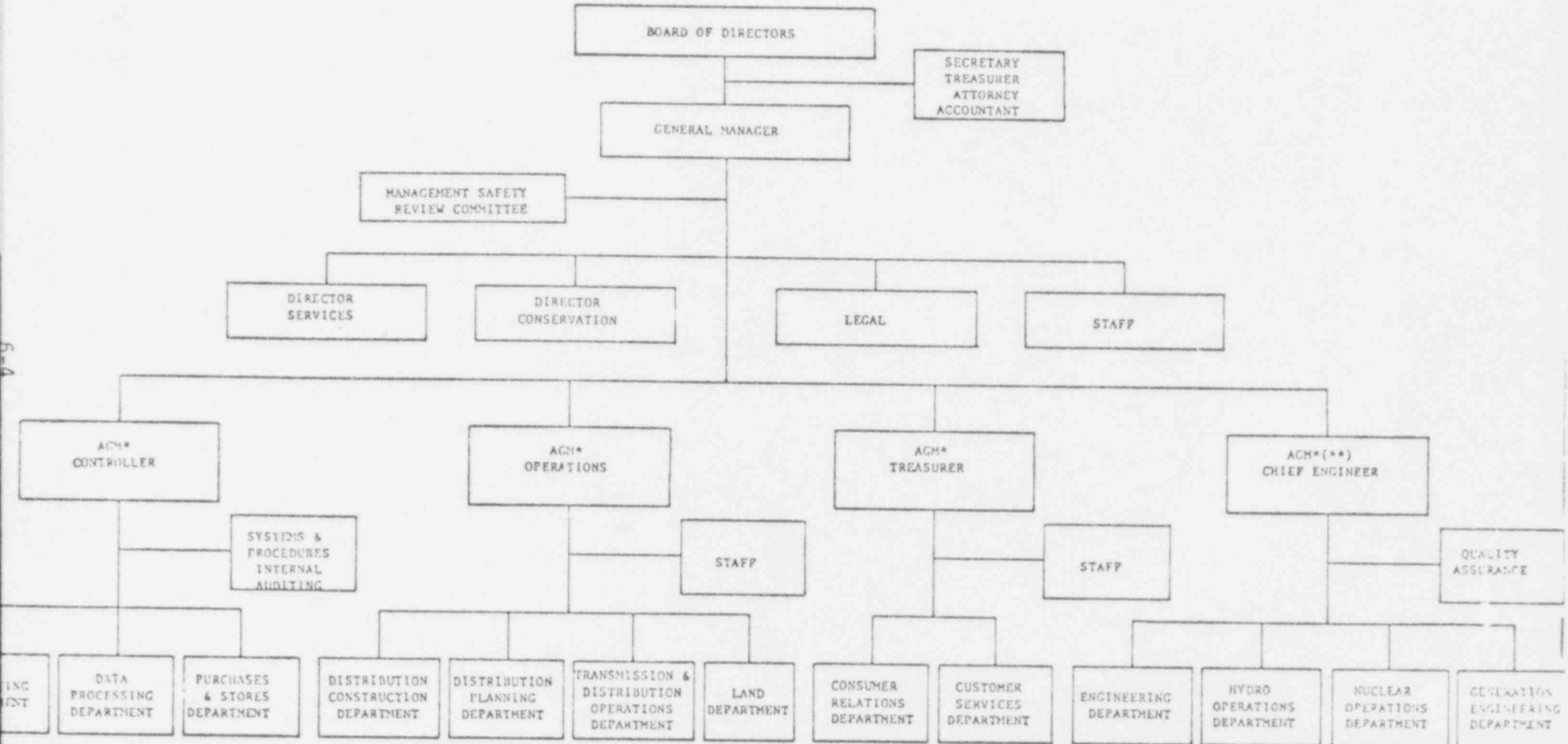
* One licensed operator when the reactor is shutdown greater than 1% $\Delta k/k$.

** In the event that any member of a minimum shift crew is absent or incapacitated due to illness or injury, a qualified replacement shall be designated to report onsite within two hours.

SL - NRC Senior Licensed Operator

L - NRC Licensed Operator

357 068



*AGM-Assistant General Manager
**Responsible for Fire Protection Program

FIGURE 6.2-1
SMUD ORGANIZATION CHART

357 069

6.5.1.6 Responsibilities (continued)

c. Review of all proposed changes to the Technical Specifications.

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

e. Investigation of all violations of the Technic.. Specifications, and shall prepare and forward a report covering evaluation and recommendations to prevent recurrence to the Plant Superintendent, Manager of Nuclear Operations, and to the Chairman of the Management Safety Review Committee.

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

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

h. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Management Safety Review Committee.

i. Review of the Plant Security Plan and implementing procedur s, and shall submit recommended changes to the Plan to the Cha.rman of the Management Safe:y Review Committee.

j. Review of every unplanned release of radioactive material to the environs; evaluate the event; specify remedial action to prevent recurrence; and document the event description, evaluation, and corrective action, and the disposition of the corrective action in the plant records.

6.5.1.7 Authority

The Plant Review Committee shall:

a. Recommend to the Plant Superintendent written approval or disapproval of items considered under 6.5.1(a) through (d) above.

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

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6.5.2.8 Audits (continued)

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

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

g. Any other area of facility operation considered appropriate by the MSRC or the General Manager.

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

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

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

m. The PROCESS CONTROL PROGRAM for the SOLIDIFICATION of radioactive wastes from liquid systems at least once per 24 months.

6.5.2.9 Authority

The MSRC shall report to and advise the General Manager on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.28.

6.5.2.10 Records

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

a. Minutes of each MSRC meeting shall be prepared, approved, and forwarded to the General Manager within 14 days following each meeting.

357 071

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. Process Control Program implementation.

g. Offsite Dose Calculation Manual implementation.

h. Effluent and environmental quality control program.

6.8.2 Each procedure and administrative policy of 6.8.1, above, and changes thereto, shall be reviewed by the PRC. Those matters pertaining to items 6.8.1a, b, c, f, and g, above, shall be approved by the Plant Superintendent prior to implementation and reviewed periodically as set forth in each document. The manager of Nuclear Operations shall approve Security Plan and Emergency Plan implementing procedures.

6.8.3 Temporary changes to procedures 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 days of implementation.

6.9 REPORTING REQUIREMENTS

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.

6.9.1. Annual Reports

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

Reports required on an annual basis shall include:

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

6.9.2 Annual Radiological Environmental Operating Report

6.9.2.1 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality

6.9.2.2 The annual radiological environmental operating reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

357 073

6.9.2.2 (continued)

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program, including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directons from one reactor; the result of land use censuses, and the results of licensee participation in the Quality Assurance Program.

6.9.3 Semi-Annual Radioactive Effluent Release Report

Routine radioactive effluent release reports covering te operating 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. The period of the first report shall begin with the data of initial criticality.

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

The radioactive effluent release reports shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter, as outlined in Regulatory Guide 1.21, with the data summarized on a quarterly basis, following the format of Appendix B thereof.

357 074

6.9.3.1(Continued)

The radioactive effluent release reports shall include an assessment of the radiation doses from radioactive effluents to individuals due to their activities inside the site boundary during the report period. All assumptions used in making these assessment (e.g., specific activity, exposure time, and location) shall be included in these reports.

The radioactive effluent release reports shall include the following information for all unplanned releases to unrestricted areas of radioactive materials in gaseous and liquid effluents:

a. A description of the event and equipment involved.

b. Cause(s) for the unplanned release.

c. Actions taken to prevent recurrence.

d. Consequences of the unplanned release.

The radioactive effluent release reports shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each calendar quarter, as outlined in Regultory Guide 1.21. In addition, the nearest offsite receptor maximum noble gas gamma air and beta air doses shall be evaluated. The meteorological conditions concurrent with the releases of effluents shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) or (ODCM) made during the reporting period, as provided in Specifications 6.14 and 6.15.

6.9.4 Monthly Report

Routine reports of operating statistics, including narrative summary of operating and shutdown experience, or major safety-related maintenance, and tabulations of facility changes (including changes to radwaste treatment system), tests or experiments required pursuant to 10 CFR 50.59(b), shall be submitted on a monthly bases to the Office of Management Information and Program Control, U. S. Nuclear Regulatory Commission, Washington, D. C., 20555, with a copy to the Regional Office, postmarked not later than the 15th day of each month following the calendar month covered by the report. In addition, any changes to the Offsite Dose Calculation Manual shall be submitted with the Monthly Operating Report within 90 days in which the change was made effective.

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6.9.5 Reportable Occurrences

The REPORTABLE OCCURRENCES of Specifications 6.9.5.1 and 6.9.5.2 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a license event report shall be completed and reference shall be made to the original report date.

6.9.5.1 Prompt Notification with Written Follow-up

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

a. Failure of the reactor protection system or other sytems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting condition for operation established in the technical specifications.

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

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

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

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6.9.5.1 Prompt Notification with Written Follow-up (Continued)

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

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

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

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

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

j. Occurrence of an unusual or important event that causes a significant environmental impact, that affects potential environmental impact from unit operation, or that has high public or potential public interest concerning environmental impact from unit operation.

k. Occurrence of radioactive material contained in gaseous holdup tanks in excess of that permitted by the limiting condition for operation established in the Technical Specifications.

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6.9.5.2 Thirty Day Written Reports

The types of events listed below shall be the subject of written reports to the Director of the Regional Office within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a Licensee Event Report form. Information provided on the Licensee Event Report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications, but which do not prevent the fulfillment of the functional requirements of affected systems.

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

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

d. Abnormal degradation of systems other than those specified in 6.9.4.1.c, above, designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of: (1) more than one curie of radioactive material in liquid effluents, (2) more than 150 curies of noble gas in gaseous effluents, or (3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

   1. A description of the event and equipment involved.

   2. Cause(s) for the unplanned release.

   3. Actions taken to prevent recurrence.

   4. Consequences of the unplanned release.

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6.9.5.2 Thirty Day Written Reports (Continued)

f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.26-2 when averaged over any calendar quarter sampling period. When more than one of the radionuclides in Table 3.26-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 3.26-2 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 3.21.2, 3.22.2, and 3.22.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

6.9.6 <u>Special Reports</u>

Special reports shall be submitted to the Director of the Regulatory Operations Regional 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 Reactor Building structural integrity report shall be submitted within 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.

b. Inservice Inspection Program

c. Reserved for Proposed Amendment No. 43.

357 079

6.9.6 Special Reports (Continued)

d.	Status of Inoperable Fire Protection Equipment.	10 days	(3.14)
e.	Radioactive Liquid Effluent Concentration	14 days	(3.21.1)
f.	Radioactive Liquid Effluent Dose	30 days	(3.21.1)
g.	Gaseous Effluents	14 days	(3.22.1)
h.	Noble Gas Limit	30 days	(3.22.2)
i.	Radioiodine and Particulates	30 days	3.22.2)
j.	Gaseous Radwaste Treatment	30 days	(3.23)
k.	Gas Storage Tanks	14 days	(3.24)
l.	Solid Radwaste System Inoperable	30 days	(3.25)
m.	Radiological Monitoring Program	30 days	(3.26)
n.	Monitoring Point Substitutions	30 days	(3.26)
o.	Land Use Census	30 days	(3.27)
p.	Fuel Cycle Dose	60 days	(3.18)

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6.10.2 (Continued)

g. Records of training and qualification for current members of the plant oprating 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 PRC and MSRC.

l. Records of radiological environmental monitoring program.

357 081

6.12 RESPIRATORY PROTECTION PROGRAM

The Respiratory Protection Program administered shall conform to the USNRC Regulatory Guide 8.15.

357 082

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Function

The PCP shall be a manual containing the equipment operating procedures, process parameters, set points, drawings and controls, and the laboratory procedures detailing the program of sampling, analysis, and evaluation within which solidification of radioactive wastes from liquid systems is assured, and the surveillance requirements of these Technical Specifications.

The PCP shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

6.14.2 Changes to the PCP shall be made by either of the following methods:

A. Licensee initiated changes:

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

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

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

c. Documentation of the fact that the change has been reviewed and found acceptable by the Plant Review Committee.

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

357 083

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 Function

The ODCM shall describe 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 consistent with the applicable LCO's contained in these Technical Specifications. Methodologies and calculational procedures acceptable to the Commission are contained in NUREG-0133.

The ODCM shall be submitted to the Commission at the time of proposed Radiological Effluent Technical Specifications and shall be subject to review and approval by the Commission prior to implementation.

6.15.2 Any changes to the ODCM shall be made by either of the following methods:

A. Licensee-initiated changes:

1. Shall be submitted to the Commission by inclusion in the Monthly Operating Report within 90 days of the date the change was made effective, and shall contain:

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

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.

357 084

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

6.16.1 Function

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

6.16.2 Major changes to the radioactive waste systems (liquid, gaseous, and solid) shall be made by either of the following methods. For the purpose of this specification, "major changes" is defined in Specification 6.16.3, below.

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

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

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

e. An evaluation of the change which shows the expected maximum exposures to individuals in the unrestricted area and to the general population from those previously estimated in the license application and amendments thereto;

357 085

6.16.2 (Continued)

f. A comparison of the predicted releases of radioactive materials in liquid and gaseous effluents and in solid waste to the actual releases for the period in which the changes were made;

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

h. Documentation of the fact that the change was reviewed and found acceptable by both the PRC and the MSRC.

2. The Change shall become effective upon review and acceptance by both the PRC and MSRC.

6.16.3 Background and definition of what constitutes "major changes" to radioactive waste systems (liquid, gaseous, and solid).

A. Background

1. 10 CFR Part 50, Section 50.34a(a) requires that each application to construct a nuclear power reactor provide a description of the equipment installed to maintain control over radioactive material in gaseous and liquid effluents produced during normal reactor operations, including operational occurrences.

2. 10 CFR Part 50, Section 50.34a (b)(2) requires that each application to construct a nuclear power reactor provide and estimate of the quantity of radionuclides expected to be released annually to unrestricted areas in liquid and gaseous effluents produced during normal reactor operation.

3. 10 CFR Part 50, Section 50.34a(3) requires that each application to construct a nuclear power reactor provide a description of the provisions for packaging, storage, and shipment offsite of solid waste containing radioactive materials resulting from treatment of gaseous and liquid effluents and from other sources.

357 086

6.16.3 (Continued)

4 10 CFR Part 50, Section 50.34a(3)(c) requires that each application to operate a nuclear power reactor shall include (1) a description of the equipment and procedures for the control of gaseous and liquid effluents and for the maintenance and use of equipment installed in radioactive waste systems, and (2) a revised estimate of the information required in (b)(2) if the expected releases and exposures differ significantly from the estimate submitted in the construction permit.

5. The Regulatory staff's Safety Evaluation Report and Amendments thereto issued prior to the issuance of an operating license contains a description of the radioactive waste systems installed in the nuclear power reactor and a detailed evaluation (including estimated releases of radioactive materials in liquid and gaseous waste and quantities of solid waste produced from normal operation, estimated annual maximum exposures to an individual in the unrestricted area and estimated exposures to the general population) which shows the capability of these system to meet the appropriate regulations.

6. The Regulatory staff's Final Environmental Statement issued prior to the issuance of an operating license contains a detailed evaluation as to the expected environmental impact from the estimated releases of radioactive material in liquid and gaseous effluents.

357 087

6.16.3 (Continued)

B. Definition

"Major Changes" to radioactive waste systems (liquid, gaseous, and solid) shall include the following:

1. Major changes in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2. Major changes in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly alter the characteristics and/or quantities of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3. Changes in system design which may invalidate the accident analysis as described in th SER (e.g., changes in tank capacity that would alter the curies released); and

4. Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of skid-mounted equipment, use of mobile processing equipment).

357 088

RANCHO SECO NUCLEAR GENERATING STATION

PROCESS CONTROL PROGRAM

May 1979

SACRAMENTO MUNICIPAL UTILITY DISTRICT

SACRAMENTO, CALIFORNIA

357 089

TABLE OF CONTENTS

357 090

1.0 PURPOSE

The purpose of this operational and technical manual (Process Control Program) for the Rancho Seco Radioactive Waste Solidification Unit is to:

1. Provide a reasonable assurance of the sati factory solidification of liquid radioactive waste and assure the absence of any significant free water.

2. Standardize the method of operation for solidification of radwaste.

3. Provide a training manual for solidification unit operations

4. Provide a system description, sample collection and analysis information, solidification tests, and acceptance criteria.

Major reference is made to the operating procedures. This is required to assure that the solid waste system is operated properly and to assure that the product of the process contains no free water and is a free standing solid.

Identified herein are: the interfaces with other plant systems (i.e., interlocks, monitors) which are required to be functional prior to radwaste solidification processing; administrative controls and equipment features to ensure that operating procedures are followed; sampling requirements prior to processing; and various processing steps and process parameters which provide boundary conditions within which the radwaste solidification system is to be operated.

Portions of this document will be incorporated into the appropriate Rancho Seco technical and training manuals as deemed appropriate for proper administrative control of the liquid radwaste solidification process.

357 0?1

2.0 REFERENCES

1. "Radiological Effluent Technical Specifications for PWR's", Section 3/4.11.3 "Solid Radioactive Waste" NUREG-0472.

2. Radiological Effluent Technical Specifications for PWR's", Section 6.14 "Process Control Program." NUREG-0472.

357 092

3.0 SOLIDIFICATION PROCESS GENERAL DESCRIPTION AND CHEMISTRY

1. General Description

1. The liquid radioactive waste is transferred to a disposable liner by P-694 (miscellaneous waste concentrate storage tank pump). Waste is added to the disposable liner until the predetermined waste level setpoint is reached. Level control on the disposable liner is maintained by a set of level leads which have an ohm meter read-out on the Control Panel H2WS1. When liquid reaches these level leads, a short is produced and is reflected on the ohm meter at Panel H2WS1. A red indication light is also located on Panel H2WS1 for each level lead.

2. Once the liquid radioactive waste has been added, the waste pump (P-694) is secured and the transfer of urea formaldehyde from the urea formaldehyde tank (T-638) is started. Urea formaldehyde is added to the disposable liner until the predetermined UF level setpoint is reached.

3. The air sparging system mixes the waste/UF and maintains a thorough mixing until the mixture is solidified.

4. Once the solution of waste and urea formaldehyde is mixed, catalyst addition is started by pressurizing the catalyst storage tank to 10 $\pm$ 2 psig. The air head forces the catalyst into the disposable liner.

2. Chemistry

The Rancho Seco in-plant liquid radwaste solidification system utilizes urea formaldehyde (UF) and sulfuric acid to convert radioactive liquid waste into a solid matrix. The urea formaldehyde is combined with the radioactive liquid and mixed. Once the UF/waste is thoroughly mixed, sulfuric acid (about 34%) is added as a catalyst to the solidification process. A strong acid (H_2SO_4) is used to minimize the amount of catalyst required to adjust the

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pH of the waste solution. The polymerization reaction that occurs during the process will result in the formation of some minimal amount of water. This free water is scavenged by the addition of a powdered, pre-catalyzed resin (Cascamite) to the liner afte the solidification process has been completed. The most important chemistry criteria for beginning the polymerization reaction is an acid pH (2-4); however, the reaction can be influenced by high solids content in the liquid waste solution, particularly high boron found in evaporator bottom samples.

The volume ratio of liquid wastes (normally evaporator bottoms) to urea formaldehyde is about 2:1. High solids content can result in less UF being required for complete polymerization. With high solids content, a 3:1 or 4:1 ratio may be sufficient. The amount of catalyst (sulfuric acid) required to reach the desired pH must be determined for each waste solution. Normally 10-25 gallons of catalyst is required for a 300 ft^3 container. A good mixture begins to gel within a minute or two and should be solidified after approximately one-half hour. Curing takes place over several hours.

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4.0 SOLIDIFICATION SYSTEM DESCRIPTION

The major systems contained in the solidification system are: the waste transfer system; the UF transfer system; the air sparging system; the catalyst addition system; the off gas vent system; the electrical control console; and, the fill head assembly. An overall view of the Rancho Seco liquid radioactive waste solidification unit is shown in SMUD Drawings M-561, Sh. 1 and Sh. 2.

The radioactive waste solidification system receives radioactive waste from Tanks T-679A and T-679B (the Miscellaneous Waste Concentrate Storage Tanks). The electrical control panel (H2WS1) is equipped with a high and low level alarm for both Miscellaneous Waste Concentrate Storage Tanks. The storage tank pump P-694 is controlled from the electrical control panel. Valve RWS-205, which is located in the solidification room, isolates the radioactive waste solidification system from the radioactive Miscellaneous Liquid Radwaste System.

Detailed systems descriptions are given below for the major components of the liquid radwaste solidification system:

1. Waste Transfer System

   .1 The purpose of the waste transfer is to transfer all types of radioactive waste to the disposable liner via a fill head assembly.

   .2 The waste transfer header is a 1-1/2 inch stainless steel header and extends from valve RWS-205 through the auxiliary building wall to a flanged connector. A waste transfer hose connects the waste transfer header to valve FV-64902 which is mounted to the fill head assembly.

   .3 The waste transfer header is equipped with a waste inlet isolation valve LV-63501 which is operated from electrical panel H2WS1 and an air operated header isolation valve (LV-64502).

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.4 A check valve (RWS-661) is installed in the waste transfer header to prevent backflushing of the solidification agent into the liquid radioactive waste system.

.5 The miscellaneous wastes concentrate tank pump (P-694) functions as the radioactive waste transfer pump for the radioactive waste solidification system.

2. Urea Formaldehyde Transfer System

.1 The purpose of the UF transfer system is to store the urea formaldehyde and transfer the urea formaldehyde to the disposable liner as required for solidification.

.2 The urea formaldehyde is stored in a urea formaldehyde storage tank, T-638, which has a storage capacity of 1,375 gals. The storage tank is equipped with internal heating elements which maintain a constant urea formaldehyde temperature of 75 F. The heaters are controlled from electric panel H2WS1. Tank level indication is also provided at the solidification control panel (H2WS1).

.3 The urea formaldehyde transfer pump (P-639) is a Moyno pump rated at 12 GPM. The transfer pump takes suction on tank T-638 (urea formaldehyde storage tank) through valve (FV-63902) and discharges to the waste transfer header through valve (FV-63903). If the discharge valve (FV-63903) is not open, the UF pump cannot be started for transfers to the liner. If the UF pump is operating and the discharge valves are shut, the UF transfer pump will be de-energized. The UF pump cannot be started unless the fill head is properly placed on the disposable liner. This is to prevent a spill of urea formaldehyde to the area outside the disposable liner.

.4 On a high-high level condition in the disposable liner, the UF pump will be de-energized. This is to prevent an overflow condition in the liner.

.5 The urea formaldehyde header is equipped with a pump suction valve (FV-63902), pump discharge valve (FV-63903), and a mini-flow recirc header.

3. <u>Catalyst Addition System</u>

.1 The purpose of the catalyst addition system is to store the catalyst and transfer the catalyst to the disposable liner as required for solidification.

.2 The catalyst is transferred to the disposable liner by pressurizing the catalyst tank (T-641) with service air. This is performed from the solid radwaste control room using service air valve SAS-472.

.3 The sparge mode is the normal method of catalyst addition. In this mode of operation, the catalyst is forced throughout the matrix by the air sparging pressure. This method ensures equal solidification throughout the mixture. Valve (FV-64904) is open and valve RWS-674 is shut.

.4 The catalyst tank is equipped with both local and remote (electrical panel H2WS1) tank level indication and the tank is also equipped with a high level alarm which is set at 34 inches in the catalyst tank (T-641).

.5 The catalyst addition system is equipped with two flush connections; one is used to flush the transfer header, the other is to flush the catalyst tank (T-641).

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4. Service Air

.1 The function of the service air system is to supply air at about 100 psig for the following functions.

  .1 Operation of air operated valves
  .2 Catalyst tank pressurization
  .3 Television camera cooling
  .4 Air sparging

5. Air Sparging Sys'em and Off Gas Vent System

.1 The purpose of the air sparging system is to mix the contents of the container and the urea formaldehyde. This is performed by bubbling air through the mixture.

.2 The major components of the air sparging system are the air sparging header which is placed inside the disposable liner and the control regulator located in the solid radwaste control room.

.3 The purpose of the off gas vent system is to create a slight vacuum at the top of the disposable liner to draw off any radioactive airborne contamination and pass them through a HEPA filter and to EF A-18 suction.

.4 During normal operation, the off gas vent blower should be operating during waste transfer, UF addition, and catalyst addition.

.5 The off gas vent system exhausts to a filtered and monitored plant exhaust stack.

6. Electrical Control Panel (H2WS1) and Interlocks

.1 Panel H2WS1 is at the solid radwaste control station. This panel contains indications and controls for the following components:

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.1 Remote viewing monitor

.2 Waste storage tanks (T-679A and B) high-low level alarms

.3 UF Tank (T-638) level and temperature indication

.4 UF Tank heater controls

.5 Catalyst tank level

.6 Jib crane controls

.7 Waste transfer pump (P-694) controls

.8 Level controls and indications

.9 Valve and fill head controls

.2 Fill Head Position Waste Transfer Valves LV-63501 and LV-63502 Interlock

.1 This interlock prevents the transfer of waste into the fill line unless the filling head is properly positioned..

.2 Valves LV-63501 and LV-63502 will not open unless the filling head is properly positioned on the disposable liner.

.3 The purpose of this interlock is to prevent inadvertant pumping of radioactive waste when the filling head is not located on a disposable liner.

.3 Waste Transfer Valves LV-63501 and LV-63502 and Blower (A-653) Interlock

.1 This interlock prevents the transfer of waste into the fill line unless blower (A-653) is running.

.2 Valves LV-63501 and LV-63502 will not open unless the blower (A-653) is running.

.3 The purpose of this interlock is to insure proper ventilation of the disposable liner while adding radioactive liquid waste.

.4 High-High Level-Waste/UF Fill Head Valve FV-64902 Interlock

.1 This interlock prevents the addition of liquid waste or UF on a high level condition in the disposable liner. Valve FV-64902 cannot be reopened until the high level condition has cleared.

.2 The Waste/UF fill head valve FV-64902 operates off a fixed position level probe.

.3 The purpose of this interlock is to prevent an overflow condition in the disposable liner.

.5 Fill Head Position-UF Transfer Pump (P-639) Outlet Valve FV-63903 Interlock

.1 This interlock prevents the opening of valve FV-63903 unless the fill head is properly positioned on the disposable liner.

.2 The purpose of this interlock is to prevent inadvertent pumping of UF unless the filling head is properly positioned on the disposable liner.

.6 High-High Level - Valve FV-64201 Interlock

.1 The interlock de-energizes relay CT4R which closes valve FV-64201 on a high level condition in the disposable liner. Valve FV-64201 cannot be re-opened until the high level condition has been cleared.

.2 It operates off a fixed position level probe. When the level reaches the probe contact, HHLAR, the opening circuit of valve FV-64201 is energized. This causes relay CT4R to de-energize shutting valve FV-64201.

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.3 The purpose of this interlock is to prevent an overflow condition of catalyst or flush water in the disposable liner.

.7 Fill Head Position- Valve FV-64201 Interlock

.1 The interlock prevents relay CT4R from being energized when the fill head is not properly positioned on the disposable liner.

.2 The interlock operates off a proximity switch located in the base of the filling head. When the fill head is properly positioned, contact FHPR in the control circuit for valve FV-64201 is closed, allowing relay CT4R to be energized thus opening valve FV-64201.

.3 This interlock is to prevent the transfer of catalyst or flush water when the fill head is not properly positioned on a disposable liner.

.8 Catalyst Tank Isolation-Catalyst Flush Interlock

.1 This interlock allows valve HV 64101 to open only when valve FV-64201 is shut.

.2 The purpose of this interlock is to prevent inadvertant addition of flush water to the catalyst tank.

.9 When a high-high level exists in the disposable liner, all flow into the liner is automatically stopped. This is to prevent a spill of radioactive waste to the surrounding environment.

.10 Panel H2WS1 is equipped with an alarm bell which will alarm on a high-high level condition in the disposable liner. An alarm cutoff switch is available to silence the alarm. When the alarm is silenced, a flashing red light is activated and will flash until the alarm condition is cleared.

.11 Panel H2WS1 is equipped with level indication for waste level, urea formaldehyde level, and high level. Waste and UF controls are adjustable and will be set at the level determined by the test solidification sample. Each level is also equippped with a red indicating light.

7. Fill Head Assembly

.1 The purpose of the fill head assembly is to allow for in-container mixing of the radioactive waste, urea formaldehyde, and catalyst. The fill head assembly also serves as a seal to prevent airborne contamination from escaping the disposable liner.

.2 The fill head assembly houses the following components:

.1 Remote monitoring camera
.2 Valve (RWS674), FV-64902, and FV-64904.
.3 Proximity switch for fill head position (ZSP64901)
.4 Pressure detector for liner pressure (PSH64901)

.3 The fill head is constructed of 3/16" mild steel and is coated with a sulfuric acid resistant paint. A neoprene gasket is fitted to the base of the fill head assembly to form a positive seal between the disposable liner and the fill head assembly. Additional protection may be provided by covering with plastic.

.4 Valve FV-64902 (fill head isolation) is a 2-inch, air-operated, direct operating, diaphragm valve.

5.0 SAMPLE COLLECTION AND ANALYSIS

This section of the Process Control Program establishes the required sampling, analysis, test solidification, and evaluation which is necessary to ensure complete solidification of liquid radioactive waste.

1. Sample Procedure Overview

   .1 Each Miscellaneous Waste Concentrate Tank will be considered a "batch". The tank will be recirculated to ensure sufficient mixing and will be isolated so additional liquid waste will not be added.

   .2 The minimum sampling requirement for test solidification is every tenth batch. However, initial sampling should be performed more frequently (each tank of liquid waste) until experience is gained to predict evaporator bottoms chemistry - solidification behavior.

   .3 If any test sample fails to solidify, a new test sample shall be obtained and alternate solidification parameters determined. When a subsequent test verifies proper solidification, then solidification of the batch may begin using the alternative solidification parameters determined.

   .4 If the first test sample from a batch of waste fails to verify solidification, a sample will be collected and analyzed in accordance with this document from each consecutive batch of the same type of wet waste until three (3) consecutive test samples demonstrate satisfactory solidification. At this point, the sampling requirement is again every tenth batch of liquid radioactive waste.

2. Radiological Precautions

.1 All samples must be handled with proper radiological considerations to minimize personnel exposure and to prevent the spread of contamination. See the Radiation Control Manual AP 305 for additional information.

.2 Cotton liners, rubber gloves, and a lab coat (as a minimum) shall be worn while collecting, handling, and testing all samples.

.3 A "clean" and "contaminated" control area should be set up to prevent contamination spread.

.4 Radiation Work Permits will control conditions and operations of sampling solidification and disposal.

.5 Completed test samples will be properly disposed of. A record of volume and description of the sample will be maintained on PCP data sheets.

3. Sampling Continuous Transfers of Waste

.1 Obtain the sample from valve RWS-691 after recirculating the tank contents until it is sufficiently mixed. This normally should be a minimum of 3 tank volumes.

.2 Sample volume should normally be 1-liter; however, if radiation levels make this impractical, a smaller sample may be obtained as appropriate.

Note: Test solidification will always be performed prior to adding any liquid radwaste into the disposable liner.

4. Sample Analysis

.1 Specific techniques for chemistry analysis are not included in this Program since there are several acceptable procedures for many of the analyses that may be required. See Chemistry and F. 'ochemistry Manual AP 306 for additional information.

.2 Evaporator bottoms should be analyzed for the following: pH, specific gravity, oil, and boron.

.3 Correlation of evaporator bottoms chemistry vs UF and catalyst required for complete solidification should be made to minimize UF and catalyst usage and establish process boundary limits.

.4 All waste should have a qualitative test for foaming upon the addition of the catalyst. This can be accomplished by the addition of the catalyst to a small quantity of the waste in a beaker and visually observing the results.

Note: High concentration or rapid addition of H_2SO_4 may cause excessive foaming.

.5 The solidification agents will require certain analysis and are included in this section for convenience.

.1 Each new shipment of UF and monthly thereafter on the storage tank, a sample should be analyzed for specific gravity and pH. Record any manufacturers lot numbers and production dates which may be available. The specifications for UF are: pH = 7.6 $\pm$ 0.2 and specific gravity = 1.296 $\pm$ 0.06. UF outside of these limits should not be used for solidification purposes.

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.2 Each shipment of catalyst, prior to use, should be analyzed for specific gravity. Any batch of catalyst that varies significantly from a specific gravity of 1.265 should not be used.

Note: 34-35% H_2SO_4 = about 1.265 specific gravity.

.6 All analytical results are to be recorded in a log maintained for that purpose.

5. Process Control Program Data Sheet

.1 The PCP data sheet will be used to collect information from each test specimen and solidification evolution. The data sheet is to be maintained on permanent file.

.2 The following information is required on each test specimen:

.1 pH of waste
.2 Waste oil content
.3 Waste/UF ratio
.4 UF/acid ratio
.5 Percentage of free standing water
.6 Amount of powdered precatalyzed resin added

.3 The following information is required for all solidification evolutions:

.1 Waste type
.2 Level probe setpoints
.3 Sparge air pressure
.4 Flow rate sparging time
.5 Total waste added
.6 Total UF added

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.7 Total catalyst added

.8 Free standing solid

.9 Batch number

.4 The batch numbers shall range from 1 through 10 and for each tenth batch, a new set of process parameters shall be determined.

.5 A PCP data sheet is included as Section 7 of this PCP. However, the data sheet need not be used in that identical form as long as the required information is permanently filed.

6.0 TEST SOLIDIFICATION AND ACCEPTANCE CRITERIA

1. General Solidification Considerations

.1 A liquid waste/UF ratio of 2:1 is to be used on the first test solidification unless previous data has shown a different ratio is normally used for liquid radioactive waste. If a 2:1 ratio results in a positive solid then increase the ratio in increments of 0.5 (waste) until test results are negative. Then use the last positive solidification result for the process ratio; however, do not exceed a waste/UF ratio of 3:1.

.2 If the pH of any waste was less then three, a caustic should be added to increase the pH to greater than three prior to the addition of the UF. Record the sample size and the amount of caustic added.

.3 If foaming occurred in the waste sample, an antifoaming agent should be added to the waste prior to the addition of the UF. Record the sample size and the amount of antifoaming agent added.

.4 If the oil analysis indicated oil concentrations greater than 1%, attempts to remove the oil should be initiated. This may be accomplished by skimming the top of the liquid or by the addition of anti-emulsion agents.

2. Test Specimen Solidification

.1 The waste sample should have the required pretreatment accomplished prior to the test solidification.

.2 Prepare the test solidification vessel (normally a 1000-ml disposable beaker) with a mixing device. This may be a disposable magnetic stirrer, a miniature air sparge system or other mechanical means of mixing the waste to UF.

.3 Transfer a known representative volume of the waste to the test solidification vessel. A typical volume is normally 400-ml.

.4 Add the appropriate amount of UF as determined by the applicable ratio. For example, 200-ml of UF would be added to 400-ml of evaporator bottoms.

.5 Initiate mixing the waste and UF and, after a homogeneous mixture is obtained (normally allow at least 10 minutes), begin the catalyst addition until a pH of approximately 2 is obtained at which time discontinue the addition of the catalyst. Record the amount of catalyst used.

.6 As soon as the mixture begins to thicken, secure the mixing and allow the sample to remain undisturbed for at least 30 minutes.

.7 If any free liquid is noted on the top of the sample, transfer the liquid, by draining, into a clean, disposable volumetric beaker and record the amount of liquid transferred. Calculate and record the percent of free liquid present. For example, if 6-ml of liquid was obtained from a 600-ml total volume, this would represent 1% free water.

.8 Add a sufficient quantity of cascamite (CR-10) to the liquid in the beaker to absorb all the liquid. Record the amount of cascamite added. The amount of cascamite added will be correlated to the amount that will be required for addition to the liner. Note and record if any liquid remains. Calculate the percent of free liquid of the entire volume of solidified waste if any liquid was present.

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3. Alternate Solidification Parameter Selection

.1 If unacceptable solidification resulted from excessive foaming, the following items should be explored to reduce subsequent foaming. Solidification testing as specified in Step 6.2 above must be repeated and results recorded.

  .1 Adding additional or different antifoaming agent.
  .2 Lowering the pH of the waste prior to the addition of the UF (the pH must remain above 3).
  .3 Reduce the addition rate or concentration of the catalyst.

4. Solidification Acceptability

.1 The final sample solidification ratio will be considered acceptable if the amount of free liquid following cascamite addition was less than 0.1%.

.2 The waste solidification will be considered acceptable from a solid mass standpoint if it is evident from its physical appearance that the solidified waste would maintain its shape if moved from the vessel. This may be determined, for example, by simply prodding with a stick or other rigid divice and observing significant resistance to penetration.

.3 If one or more of the above tests (4.1 and 4.2) fails to meet the stated criteria, additional solidification parameters must be determined. This will also require the initiation of the additional solidification testing requirements of three consecutive batches to show proper solidification per 6.1.1.

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7.0 PCP DATA SHEET

Tank No. ________
Length of recirculation (hrs.) ________
Date ________

Test Specimen			Solidification Process		
pH		________	Waste Type		________
Waste Oil Content		________	Level Probe Setpoints	Waste	________
				UF	________
				High-high	________
Waste/UF Ratio		________	Sparge Air Pressure		________
UF/Acid Ratio		________	Flow Rate Sparging Time (Minutes)		________
Specific Gravity		________	Total Waste Added per Liner		________
			Total Catalyst Added per Liner		________
Boron		________	Free Standing Solid	Yes	________
				No	________
Foaming Positive		________	Liner Size		________
Foaming Negative		________	Amount of Pre-Catalyzed Powdered Resin Added		________
Amount of Pre-Catalyzed Powdered Resin Added		________	Batch No.		________
% Free Liquid		________			
Free Standing in Solid	Yes	________			
	No	________			
Batch Acceptable	Yes	________			
	No	________			

Operator Signature ____________________ (date)

Sr. Chem Rad. Asst. ____________________ (date)

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