ATTACHMENT 1

Proposed Technical Specification Amendment No. 182, Revision 2

Permanently Defueled Technical Specification

Appendix C to Facility Operating License No. DPR-54

APPENDIX C

TO

FACILITY OPERATING LICENSE NO. DPR-54

TECHNICAL SPECIFICATIONS

FOR THE

DEFUELED RANCHO SECO FACILITY

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DEFINITIONS

The following terms are defined for uniform interpretation of these technical specifications.

D1.1 ACTION

ACTION including time requirements shall be that part of a specification which prescribes remedial measures required under designated conditions.

D1.2 PERMANENTLY DEFUELED MODE

The plant is in a PERMANENTLY DEFUELED MODE when no fuel is in the Reactor Building and the fuel remaining on site is stored in the spent fuel pool. The PERMANENTLY DEFUELED MODE reflects the District's intent to safely store spent fuel but not operate the reactor or move fuel into the Reactor Building.

D1.3 OPERABLE - OPERABILITY

A component or system is OPERABLE when it is capable of performing its intended function within the required range. The component or system shall be considered to have this capability when: (1) it satisfies the Limiting Conditions for Operation defined in Section D3, (2) it has been tested periodically in accordance with Section D4 and has met its performance requirements, (3) the system has available its source of power, and (4) its required auxiliaries are capable of performing their intended function.

D1.4 INSTRUMENTATION SURVEILLANCE

D1.4.1 INSTRUMENT CHANNEL

An INSTRUMENT CHANNEL is the combination of sensor, wires, amplifiers, and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control, and/or protection. An INSTRUMENT CHANNEL may be either analog or digital.

D1.4.2 CHANNEL TEST

A CHANNEL TEST is the injection of an internal or external test signal into the channel to verify its proper response, including alarm and/or trip initiating action, where applicable.

D1.4.3 INSTRUMENT CHANNEL CHECK

An INSTRUMENT CHANNEL CHECK is a verification of acceptable instrument performance by observation of its behavior and/or state. This verification includes, where possible, comparison of output and/or state of independent channels measuring the same variable.

D1.4.4 INSTRUMENT CHANNEL CALIBRATION

An INSTRUMENT CHANNEL CALIBRATION is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. INSTRUMENT CHANNEL CALIBRATION shall encompass the entire channel, including equipment actuation, alarm, or trip, whichever are applicable, and shall be deemed to include the CHANNEL TEST.

D1.4.5 FUNCTIONAL TEST

A FUNCTIONAL TEST shall be the determination or verification of the capability of a system or component to meet specified requirements by subjecting the system or component to a set of physical or operating conditions.

D1.5 SITE BOUNDARY

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

D1.6 PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

D1.7 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents and used in the calculations of gaseous and liquid effluent monitoring Alarm/Trip Setpoints. The ODCM shall also contain the Radioactive Effluent Controls Program required by Specification D6.8.3a and descriptions of the information that should be included in the Semiannual Radioactive Effluent Release Report required by Specification D6.9.2.

D1.8 MEMBER(S) OF THE PUBLIC

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

D1.9 LICENSEE EVENT REPORT

Defined under Administrative Controls Specification D6.9.5.

D1.10 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL

The RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL shall contain a description of the Rancho Seco radiological environmental monitoring program. The REMP MANUAL shall also contain the REMP requirements of Specification D6.8.3b, a description of the environmental samples to be collected, sample locations, sampling frequencies, sample analysis criteria, and a description of the information to be included in the Annual Radiological Environmental Operating Report as required by Specification D6.9.1.3.

D1.11 NUCLEAR SAFETY

NUCLEAR SAFETY shall refer to those systems, components, and administrative controls that have or may have an effect on the health and safety of the public.

D1.12 INDUSTRIAL AREA

That portion of the site property, the access to which is controlled as described in the NRC approved Security Plan by security fencing, equipment, and personnel.

D1.13 UNRESTRICTED AREA

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

TABLE D1-1*

SURVEILLANCE INTERVALS

Frequency	Notation	Definition
SHIFTLY	S	At least once per 12 hours.
DAILY	D	At least once per 24 hours.
WEEKLY	W	At least once per 7 days.
MONTHLY	M	At least once per 31 days.
QUARTERLY	Q	At least once per 92 days.
6 MONTHS	SY	At least once per 184 days.
ANNUALLY	A	At least once per 12 months.
18 MONTHS	18M	At least once per 18 months.

* See Specification D4.0.2.

D3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

D3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

D3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the PERMANENTLY DEFUELED MODE, except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

D3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION including time requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION is not required.

SURVEILLANCE REQUIREMENTS

D4.0.1 Surveillance Requirements shall be met during the PERMANENTLY DEFUELED MODE or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

D4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

D4.0.3 Failure to perform a Surveillance Requirement within the specified surveillance interval, as defined in Specification D4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits for ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. An ACTION requirement may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

D4.0.4 Entry into a specified condition (i.e., fuel handling operations) shall not be made unless the Surveillance Requirement(s) associated with a Limiting Condition of Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to conditions as required to comply with ACTION requirements.

D3/4.1 SPENT FUEL POOL LEVEL

LIMITING CONDITIONS FOR OPERATION

D3.1.1 At least 37 feet of water shall be maintained in the spent fuel pool during fuel handling operations.

D3.1.2 The water level in the spent fuel pool may be less than 37 feet when fuel handling operations are not in progress as long as:

a. the measured dose rate at the surface of the pool from irradiated fuel and control components seated in the pool is ≤2.5 mRem/hr, and

b. the pool bulk coolant temperature is within the limits of Specification D3.2;

Otherwise, maintain at least 37 feet of water in the spent fuel pool.

Applicability:

Whenever irradiated fuel assemblies and control components are in the spent fuel pool.

Action:

With the requirements of LCO D3.1.1 not satisfied, suspend fuel handling operations, place fuel in a safe condition, and take immediate actions to restore the spent fuel pool level to within its limit.

With the requirements of LCO D3.1.2 not satisfied, restore the spent fuel pool level to within its limits.

SURVEILLANCE REQUIREMENTS

D4.1.1 DAILY verify the water level in the spent fuel pool.

D4.1.2 DAILY verify the spent fuel pool surface dose rate is ≤2.5 mRem/hr and the requirements of Specification D3/4.2 are met when fuel handling operations are not in progress and the spent fuel pool level is <37 feet.

D4.1.3 At least once every 18 MONTHS, perform an INSTRUMENT CHANNEL CALIBRATION on the spent fuel pool level alarm switches.

D3/4.2 SPENT FUEL POOL TEMPERATURE

LIMITING CONDITIONS FOR OPERATION

D3.2 Maintain the spent fuel pool bulk coolant temperature below 180°F.

Applicability:

Whenever irradiated fuel assemblies are in the spent fuel pool.

Action:

a. Place an alternate cooling method in service as a supplement to the spent fuel pool cooling system prior to exceeding a bulk coolant temperature of 180°F.

b. With the spent fuel pool cooling system inoperable, place an alternate cooling method in service if the pool bulk coolant temperature reaches 140°F.

SURVEILLANCE REQUIREMENTS

D4.2 DAILY verify the spent fuel pool bulk coolant temperature.

D3/4.3 FUEL STORAGE BUILDING LOAD HANDLING LIMITS

LIMITING CONDITIONS FOR OPERATION

D3.3 No loads shall be handled over irradiated fuel assemblies stored in the spent fuel pool, except fuel assemblies and control components.

APPLICABILITY:

Whenever irradiated fuel assemblies are stored in the spent fuel pool.

ACTION:

With the above requirements not met, place the fuel storage building fuel handling bridge and overhead crane in a safe condition.

SURVEILLANCE REQUIREMENTS

D4.3.1 Perform a dead weight load test at the rated load on the fuel storage building fuel handling bridge and overhead crane within 7 days prior to initial use during fuel handling operations.

D4.3.2 Perform a FUNCTIONAL TEST of the fuel storage building fuel handling bridge interlocks within 7 days prior to the commencement of fuel handling operations and once per 7 days thereafter during fuel handling operations.

D3/4.4 SPENT FUEL STORAGE AREA RADIATION MONITOR

LIMITING CONDITIONS FOR OPERATION

D3.4 Radiation levels in the spent fuel storage area shall be monitored by a fixed radiation monitor.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With fixed radiation monitoring equipment inoperable, suspend fuel handling operations, place fuel in a safe condition, initiate actions to return a fixed radiation monitor to OPERABLE status, and place portable radiation survey instrumentation into service that has the appropriate ranges and sensitivity to protect individuals in the spent fuel storage area until a fixed radiation monitor is returned to OPERABLE status.

If a fixed radiation monitor is not returned to OPERABLE status within 7 days, prepare and submit a report within an additional 30 days to the NRC Regional Administrator, Region V documenting the condition along with a plan to return a fixed radiation monitor to OPERABLE status.

SURVEILLANCE REQUIREMENTS

D4.4.1 SHIFTLY during fuel handling operations, or WEEKLY otherwise, perform an INSTRUMENT CHANNEL CHECK.

D4.4.2 MONTHLY perform an INSTRUMENT CHANNEL TEST.

D4.4.3 QUARTERLY perform an INSTRUMENT CHANNEL CALIBRATION.

D3/4.5 SPENT FUEL POOL WATER CHEMISTRY

LIMITING CONDITIONS FOR OPERATION

D3.5 The spent fuel pool water chemistry shall be maintained within the limits specified in Table D3.5-1.

APPLICABILITY:

AT ALL TIMES

ACTION:

With a water chemistry limit exceeded, initiate actions to restore the water chemistry to within limits and conduct an evaluation to determine the cause.

SURVEILLANCE REQUIREMENTS

D4.5 The spent fuel pool water chemistry shall be determined to be within the limits by analysis for the parameters and at the frequencies specified in Table D3.5-1.

TABLE D3.5-1

SPENT FUEL POOL WATER CHEMISTRY

Parameter	Units	Limit	Analysis Frequency
Chloride	ppm	≤0.15	MONTHLY
Fluoride	ppm	≤0.15	MONTHLY

D3/4.6 LIQUID HOLD-UP TANKS

LIMITING CONDITIONS FOR OPERATION

D3.6 The quantity of radioactive material contained in each of the following tanks shall be limited to $\leq$10 Curies, excluding Tritium and dissolved or entrained noble gases:

a. A or B Regenerant Hold-up Tanks

b. Demineralized Reactor Coolant Storage Tank

c. Miscellaneous Water Hold-up Tank

d. Borated Water Storage Tank

e. Outside Temporary Tanks

APPLICABILITY:

AT ALL TIMES

ACTION:

With the quantity of radioactive material in any of the listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, and initiate actions to reduce the tank's contents to within the limit. Describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report pursuant to Specification D6.9.2.

SURVEILLANCE REQUIREMENTS

D4.6 The quantity of radioactive material contained in each tank listed in Specification D3.6 shall be determined to be within the specified limit by analyzing a representative sample of the tank's contents at least WEEKLY when radioactive materials are being added to the tank.

D3/4.7 SEALED SOURCE LEAK TESTING

LIMITING CONDITIONS FOR OPERATION

D3.7 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: AT ALL TIMES.

ACTION:

With a sealed source having removable contamination in excess of an above limit, immediately withdraw the sealed source from use and:

a. Either decontaminate and repair the sealed source, or

b. Dispose of the sealed source in accordance with NRC Regulations.

SURVEILLANCE REQUIREMENTS

D4.7.1 Test Requirements

Each sealed source shall be tested for leakage and/or contamination by:

a. The Licensee, or

b. Other persons specifically authorized by the NRC, or

c. An Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

D4.7.2 Test Frequencies

Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. For sources in use at least once per 6 MONTHS for all sealed sources containing radioactive material:

    1. With a half-life greater than 30 days (excluding Tritium), and

    2. In any form other than gas.

b. For stored sources not in use each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 MONTHS. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

RANCHO SECO UNIT 1
PERMANENTLY DEFUELED
TECHNICAL SPECIFICATIONS

BASES

FOR

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

FOR THE

PERMANENTLY DEFUELED TECHNICAL SPECIFICATIONS

NOTE

The BASES contained in the succeeding pages summarize the reasons for the Specifications in Section D3/4, but in accordance with 10 CFR 50.36, are not part of these Technical Specifications.

D3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

D3/4.0 APPLICABILITY

BASES

Specification D3.0.1 and D3.0.2 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in 10 CFR 50.36(c)(2).

Specification D3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in the PERMANENTLY DEFUELED MODE or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, the facility is to be placed into a condition in which the specification no longer applies.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When the facility is placed into a specified condition in order to comply with ACTION requirements, the plant may have entered a condition in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification D3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the

D3/4.0 APPLICABILITY

BASES (Con't)

remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specifications D4.0.1 through D4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification D4.0.1 establishes the requirement that surveillances must be performed during the PERMANENTLY DEFUELED MODE or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in a condition for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

Specification D4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified. The limitation of specification D4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification D4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification D4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are

D3/4.0 APPLICABILITY

BASES (Con't)

OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification D4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification D4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification D4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the ACTION requirements are less than 24 hours, a 24-hour allowance is provided to permit a delay in implementing the ACTION requirements. This provides an adequate time limit to complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification D4.0.4 establishes the requirement that all applicable surveillances must be met before entry into the condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into the condition for which these systems and components ensure safe operation of the facility.

D3/4.1 SPENT FUEL POOL LEVEL

BASES

Maintaining the spent fuel pool water level at a minimum of 37 feet during fuel handling operations ensures a minimum of 9 feet of water over the active fuel line of spent fuel assemblies during their movement. This ensures adequate shielding is provided during fuel handling operations. This condition is analyzed and described in Updated Safety Analysis Report (USAR) Section 9.8.2. Several years of operating experience has shown this limit is adequate to protect fuel handling personnel and the health and safety of the public when fuel handling operations are in progress.

The surface dose rate and bulk coolant temperature limits ensure that adequate shielding and pool cooling are provided to ensure plant personnel safety and public health and safety when fuel handling operations are not in progress and the pool level is less than 37 feet. This condition is analyzed and described in USAR Section 9.8.2. Several years of operating experience has shown these limits are adequate to protect plant personnel and the health and safety of the public when fuel handling operations are not in progress.

D3/4.2 SPENT FUEL POOL TEMPERATURE

BASES

Limiting the spent fuel pool bulk coolant temperature to 180°F achieves the design objective of preventing spent fuel pool boiling and ensures plant personnel safety and public health and safety during the PERMANENTLY DEFUELED MODE.

If the SFP cooling system becomes inoperable or fails to maintain the SFP temperature below 140°F, one train of the Decay Heat Removal System (DHRS) is available for service to ensure the SFP bulk coolant temperature will not reach the boiling point. The intent of maintaining one train of DHRS available as an alternate means of cooling is to supplement or supplant, as necessary, the SFP cooling system until the SFP cooling system is repaired.

Use of one train of DHRS as a backup to the SFP cooling system has been previously evaluated and approved for use at Rancho Seco by the NRC in Amendment No. 84 to operating license DPR-54, and determined adequate to prevent SFP boiling. The 'A' train of the DHRS is administratively maintained available and can be lined up to provide SFP cooling per the appropriate operating procedures.

D3/4.3 FUEL STORAGE BUILDING LOAD HANDLING LIMITS

BASES

The handling of loads in the fuel storage building has been previously evaluated by the NRC and is described and analyzed in USAR Section 9.8.2.5. This heavy load specification meets the guidelines established in NUREG-0612 for the control of heavy loads at nuclear power plants and provides reasonable assurance that no heavy loads will be handled over irradiated fuel. Therefore, the potential for accidental load drops over spent fuel is minimized. Several years of operating experience has shown that these requirements are sufficient to protect the safety of plant personnel and the health and safety of the public. Retaining the existing licensing basis for Rancho Seco in the PDTS provides and maintains an acceptable level of safety in the PDM.

The restriction on the movement of loads over fuel assemblies in the spent fuel pool to fuel assemblies and control components provides reasonable assurance that in the event a load is dropped, (1) the activity released will be limited to that contained in a single fuel assembly, and (2) the distortion of fuel in the spent fuel pool storage racks will not result in a critical array. These assumptions are consistent with the accident analysis for a dropped fuel assembly.

D3/4.4 SPENT FUEL STORAGE AREA RADIATION MONITOR

BASES

This Specification satisfies 10 CFR 70.24, Criticality Accident Requirements, which mandates the need for a detection and alarm system in any area containing special nuclear material in excess of certain specified amounts. The use of radiation monitoring equipment in the spent fuel storage area ensures that early notification of excessively high radiation is provided to protect individuals involved in fuel handling operations and other activities. A less conservative form of these spent fuel storage area requirements was evaluated and approved for use at Rancho Seco by the NRC as part of the original licensing basis of the plant. Retaining and enhancing the licensing basis that existed for Rancho Seco as an operating nuclear power plant provides and maintains an acceptable level of safety in the PERMANENTLY DEFUELED MODE. Also, the requirements contained in this Specification are consistent with the Standard Technical Specifications, including the use of portable radiation survey instrumentation when fixed radiation monitoring equipment is inoperable. Several years of operating experience has proven these requirements are adequate to protect the safety of plant personnel and the health and safety of the public.

D3/4.5 SPENT FUEL POOL WATER CHEMISTRY

BASES

Specification D3/4.5 is adapted from the Appendix A Technical Specification water chemistry requirements for the reactor coolant system. The basis for these requirements is derived from the B&W Water Chemistry Manual - Dual Type Plants, BAW-1385, April 24, 1974. Fifteen years of plant operational history proves these limits are adequate to protect SFP components from significant degradation.

Excessive levels of chloride can contribute to the stress corrosion cracking of stainless steel. Also, excessive levels of fluoride can cause accelerated localized corrosion and hydriding of zircaloy. The following SFP related components contain material that is subject to potential degradation:

1. The spent fuel pool is lined with stainless steel.
2. The fuel storage racks are made of stainless steel.
3. The fuel assemblies contain a small quantity of stainless steel.
4. Zircaloy is used as cladding for the fuel in the fuel assemblies.

Therefore, maintaining the spent fuel pool water chemistry within the specified limits ensures the fuel assemblies, the spent fuel pool liner, and the fuel storage racks are protected against potential degradation.

D3/4.6 LIQUID HOLD-UP TANKS

BASES

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

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

D3/4.7 SEALED SOURCE LEAK TESTING

BASES

This Specification meets the requirements of 10 CFR 39.35, Leak Testing of Sealed Sources, and ensures leakage from byproduct, source, and special nuclear radioactive mat ial sources will not exceed allowable intake values. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on the 10 CFR 70.39(c) limits for Plutonium. 10 CFR 39.35(b) requires that sealed source leak testing must be performed using a method approved by the NRC or an Agreement State.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

DESIGN FEATURES

D5.1 SITE

The Rancho Seco site is located on the 2,480 acres owned by Sacramento Municipal Utility District, 26 miles north-northeast of Stockton and 25 miles southeast of the City of Sacramento, California. USAR Figure 1.1-2 shows the plan of the site. The Emergency Planning Zone for the PERMANENTLY DEFUELED MODE is the INDUSTRIAL AREA.

D5.1.1 Emergency Planning Zone

The Emergency Planning Zone is shown in Figure D5.1-1. It is the same as the Industrial Area.

D5.1.2 Site boundary For Gaseous Effluent

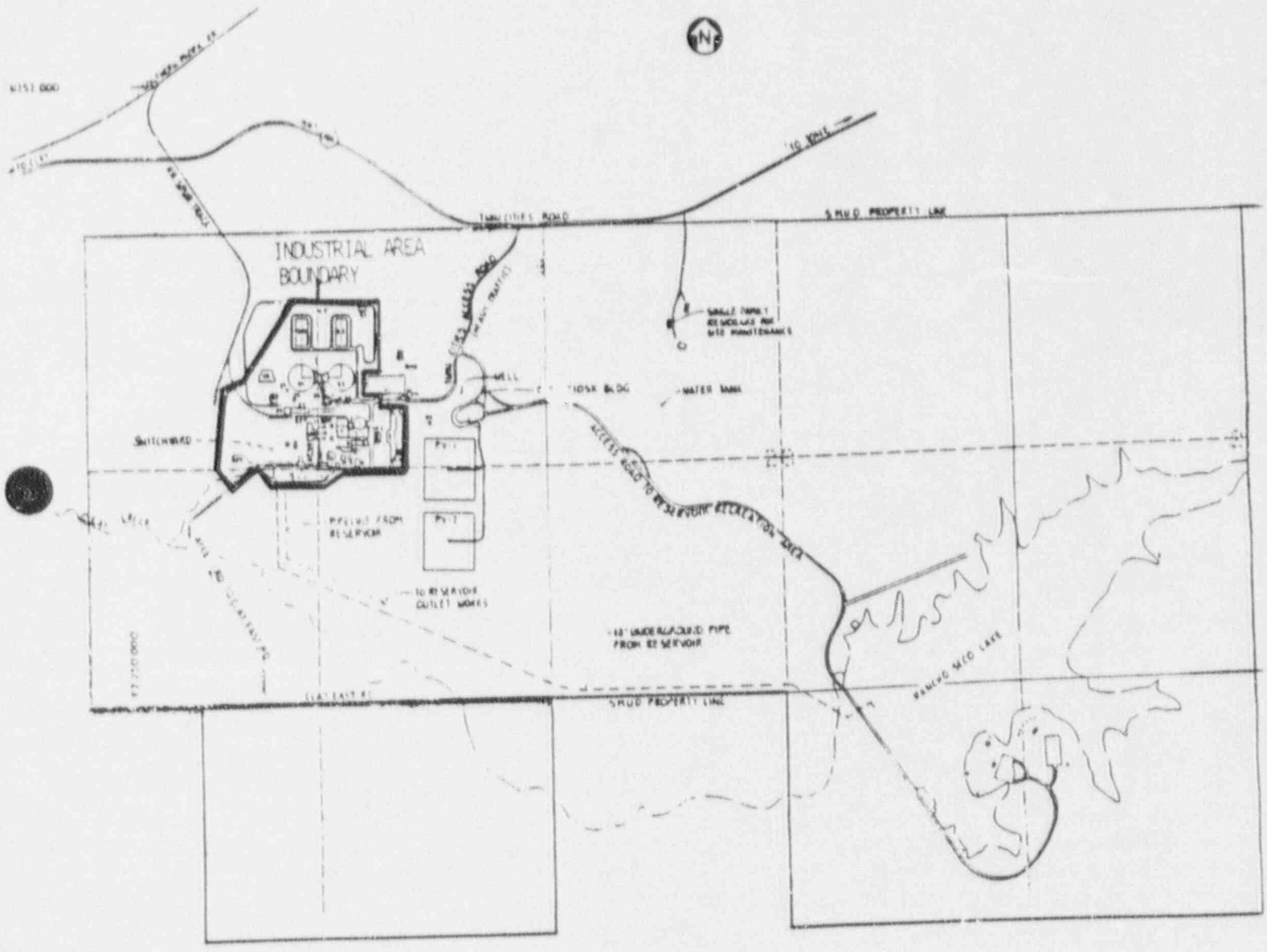
The Site Boundary for Gaseous Effluent for 10 CFR 20 compliance and for meeting 10 CFR 50, Appendix I guidelines is shown in Figure D5.1-2.

D5.1.3 Boundary For Liquid Effluent

The boundary for Liquid Effluent for 10 CFR 20 Compliance is shown in Figure D5.1-3. The dose accountability points for meeting the 10 CFR 50, Appendix I guidelines remains at the A and B Regenerant Hold-Up Tanks. Concentration accountability points for determining 10 CFR 20, Appendix B compliance remains at the Retention Basins.

FIGURE D5.1-1

EMERGENCY PLANNING ZONE



Proposed
Amendment No. 182, Revision 2

RANCHO SECO UNIT 1
PERMANENTLY DEFUELED
TECHNICAL SPECIFICATIONS

FIGURE D5.1-2

SITE BOUNDARY FOR GASEOUS EFFLUENT

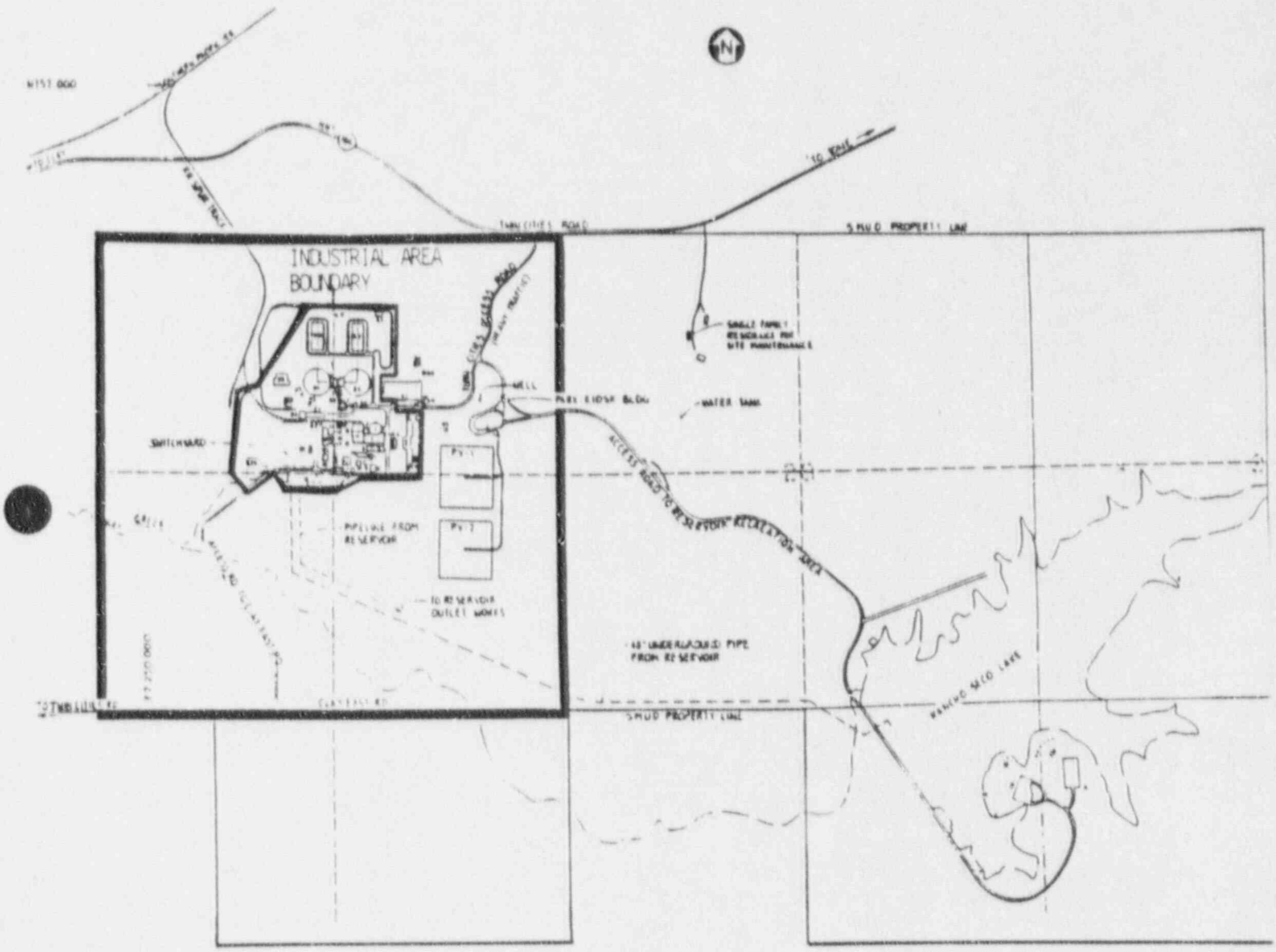


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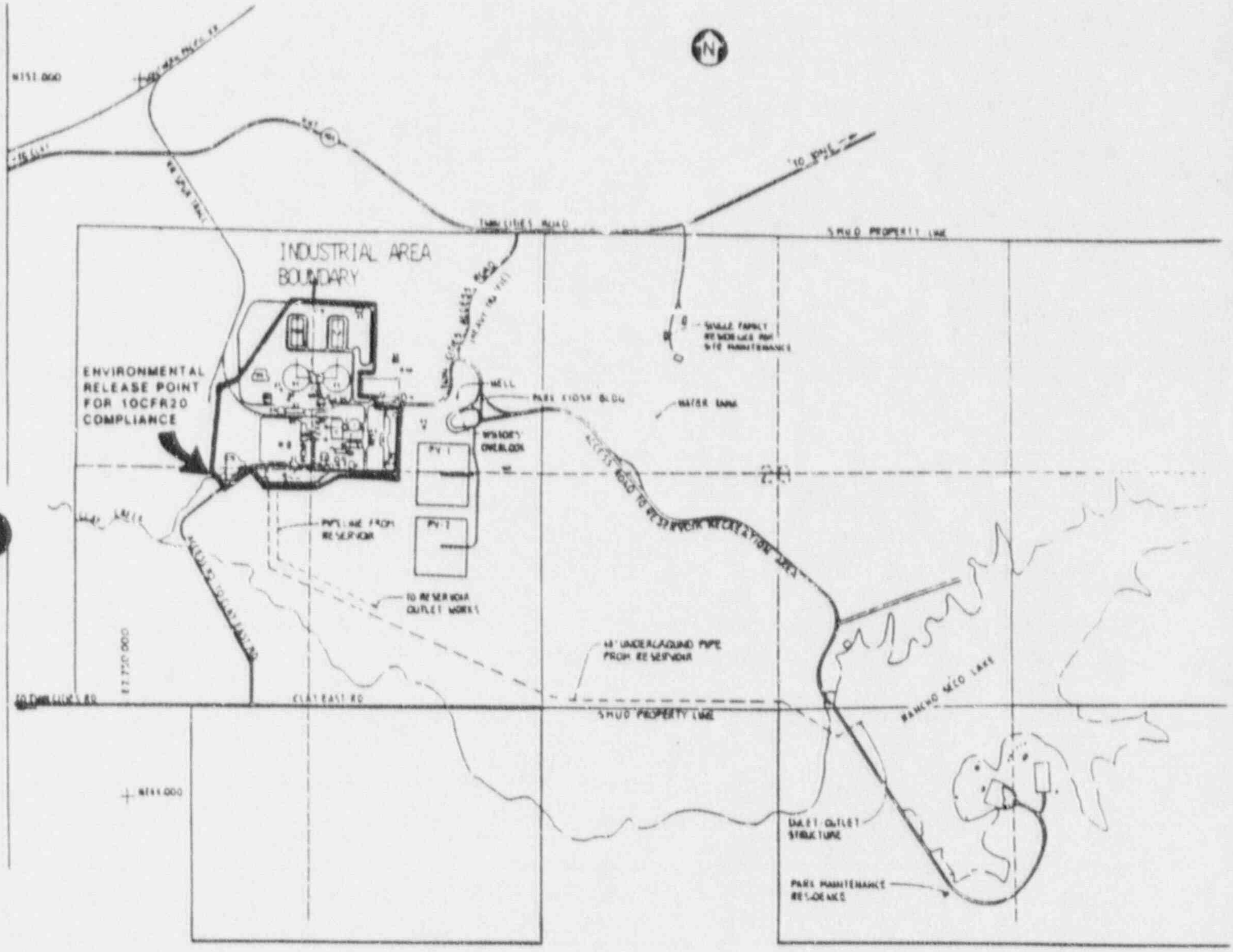
FIGURE D5.1-3

BOUNDARY FOR LIQUID EFFLUENT



D5.2 SPENT FUEL STORAGE FACILITIES

D5.2.1 Spent Fuel Storage Racks and Failed Fuel Storage Container Rack

Irradiated or failed fuel shall be stored in the stainless steel lined spent fuel pool prior to offsite shipment. The spent fuel pool is sized to accommodate 1080 fuel assemblies, plus four assemblies in failed fuel containers.

The pool has the capability of storing fuel assemblies in eleven freestanding stainless steel rack modules and four failed fuel assemblies in a special rack module. All assemblies are on nominal 10.5 inch centers in both directions. This spacing with the neutron absorber material is sufficient to maintain K_{eff} <0.95 when flooded with unborated water, based on a fuel enrichment of 4.0 weight percent.

D5.2.2 Spent Fuel Pool and Storage Rack Design

The spent fuel pool and all storage racks are designed for the plant Design Basis Earthquake.

ADMINISTRATIVE CONTROLS

D6.1 RESPONSIBILITY

D6.1.1 The Assistant General Manager (AGM), Nuclear shall be responsible for the management of the overall facility, the operation and maintenance of the plant, and the safe storage of irradiated core components, and shall delegate in writing the succession of his responsibilities during absences.

D6.1.2 The Shift Supervisor or a designated individual who is a Certified Fuel Handler shall be responsible for the control room command function. A management directive to that effect, signed by the AGM, Nuclear, shall be issued to all plant personnel on an ANNUAL basis.

D6.2 ORGANIZATION

ONSITE AND CORPORATE ORGANIZATIONS

D6.2.1 Corporate and onsite organizations shall be established for corporate management and facility operation, respectively, during the PERMANENTLY DEFUELED MODE. The organizations shall include the positions responsible for activities affecting the safe storage of irradiated core components and the overall safety of the facility.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels, to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. The organization and responsibilities shall be documented in the USAR.

b. The General Manager shall have corporate responsibility for the overall safe operation of the facility and shall take any measures necessary to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the facility to ensure NUCLEAR SAFETY.

c. The individuals who train the Certified Fuel Handlers and those who carry out health physics functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

d. The Quality Assurance organization shall report to the AGM, Nuclear to ensure independence from operating pressures.

ADMINISTRATIVE CONTROLS

FACILITY STAFF

D6.2.2 The facility organization shall be as shown in the USAR, and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table D6.2-1.

b. At least one qualified Control Room operator shall be in the Control Room when fuel is in the Spent Fuel Pool.

c. An individual qualified in radiation protection practices and procedures shall be on site when fuel is in the Spent Fuel Pool.

d. All fuel handling operations shall be directly supervised by a Certified Fuel Handler.

e. The individual who supervises the Shift Supervisors shall be a Certified Fuel Handler.

TABLE D6.2-1

MINIMUM SHIFT CREW REQUIREMENTS*

Position	Defueled
Shift Supervisor**	1
Non-Certified Operator	1
Minimum Total Personnel	2

* - 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 on site within 2 hours.

** - The Shift Supervisor shall be a Certified Fuel Handler.

ADMINISTRATIVE CONTROLS

D6.3 FACILITY STAFF QUALIFICATIONS

Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

D6.4 TRAINING

Retraining and replacement training programs for the Certified Fuel Handlers shall be maintained under the direction of the AGM, Nuclear and shall be conducted in accordance with the NRC approved training programs.

D6.5 REVIEW AND AUDIT

D6.5.1 Nuclear Safety Review and Audit Committee (NSRAC)

FUNCTION

D6.5.1.1 The NSRAC shall provide independent review and audit on matters related to:

a. NUCLEAR SAFETY,
b. facility operations in the PERMANENTLY DEFUELED MODE,
c. nuclear engineering,
d. chemistry and radiochemistry,
e. metallurgy,
f. instrumentation and control,
g. radiological safety,
h. mechanical and electrical engineering, and
i. quality assurance practices.

The NSRAC shall advise the AGM, Nuclear on these matters.

COMPOSITION

D6.5.1.2 The NSRAC shall be comprised of a Chairman and a minimum of six members. The NSRAC membership shall be made up of individuals filling positions in the Nuclear Organization who meet or exceed the minimum qualifications of ANSI N18.1-1971, Section 4.2 or 4.4, and at least 2 individuals are from outside the line nuclear organization who meet or exceed the minimum qualifications of ANSI/ANS 3.1-1981, Section 4.7.2.

ADMINISTRATIVE CONTROLS

MEMBERSHIP

D6.5.1.3 The Chairman of the NSRAC shall be the Deputy AGM, Nuclear. A current list of members of the NSRAC shall be developed and approved by the AGM, Nuclear and maintained by the NSRAC secretary. In the absence of the NSRAC Chairman, an alternate Chairman may be designated by the NSRAC Chairman or the AGM, Nuclear. The alternate Chairman shall be selected from the current list of NSRAC members.

D6.5.1.4 A list of alternate NSRAC members shall be developed and approved by the AGM, Nuclear and maintained by the NSRAC secretary.

MEETING FREQUENCY

D6.5.1.5 The NSRAC shall meet at least once every other calendar month, as convened by the NSRAC Chairman, or more frequently as directed by the AGM, Nuclear.

QUORUM

D6.5.1.6 A quorum of the NSRAC shall consist of a majority of the current NSRAC members, including the Chairman or alternate Chairman. No more than two alternates shall participate in NSRAC activities at any one time for the purpose of establishing a quorum.

RESPONSIBILITIES

D6.5.1.7 The NSRAC shall be responsible for review of:

a. The safety evaluations performed on proposed changes, tests, or experiments under the provisions of 10 CFR 50.59 that affect NUCLEAR SAFETY.

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

c. Proposed changes to the Technical Specifications or the Facility Operating License.

- NOTE -

Changes, tests and experiments, which are determined by a Qualified Reviewer and a second level review to not involve (1) an unreviewed Safety Question, (2) a change to the Facility License or Technical Specifications, or (3) a change to a licensing basis document, are not required to be reviewed by the NSRAC, except when otherwise stated in a technical specification.

d. The safety evaluations for (1) all procedures, plans, manuals, and programs required by Specification D6.8 and changes thereto, and (2) any other procedures, plans, manuals, and programs or changes thereto which are determined by the AGM, Nuclear to affect NUCLEAR SAFETY.

e. Investigations of violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, internal procedures or internal instructions having NUCLEAR SAFETY significance.

f. Significant deviations from normal and expected performance of plant equipment that affect NUCLEAR SAFETY or represent a potential safety hazard.

g. LICENSEE EVENT REPORTS as defined by 10 CFR 20.405, 50.73, and 73.71, and NUREG-1022 to determine adequacy of corrective action and to detect a degrading trend.

h. Any special investigation or facility activity brought to the attention of the NSRAC by the District's executive management which may be indicative of conditions adverse to NUCLEAR SAFETY.

i. Major changes to Radioactive Waste Treatment Systems (Liquid, Gaseous and Solid).

j. Any accidental, unplanned, or uncontrolled release of radioactive material to the environs, including the reports covering the evaluation, recommendations, and corrective actions taken to prevent recurrence. The NSRAC shall forward these reports to the AGM, Nuclear.

D6.5.1.8 The NSRAC shall provide oversight review of:

a. the Quality Assurance Program, including the Audit Program content, implementation, and reports,

b. the Radioactive Solid Waste Program and Radioactive Effluent Controls Program, and

c. the effectiveness of the LICENSEE EVENT REPORT Program per Specification D6.6.

AUTHORITY

D6.5.1.9 The NSRAC shall:

a. Report to and advise the AGM, Nuclear of activities in those areas of responsibility specified in Specification D6.5.1.7 and D6.5.1.8.

b. Render determinations in writing with regard to whether or not each item considered under Specifications D6.5.1.7a through 7e, and 7i above constitutes an unreviewed safety question.

ADMINISTRATIVE CONTROLS

c. Recommend to the AGM, Nuclear other areas of facility activities where additional auditing is prudent.

d. Advise the AGM, Nuclear of the need for independent auditing of facility activities.

D6.5.1.10 The NSRAC may form subcommittees to screen reviews required by Specification D6.5.1.7.

RECORDS

D6.5.1.11 Minutes of each NSRAC meeting, including appropriate documentation of reviews performed per Specifications D6.5.1.7c, e, f, and g, shall be prepared and forwarded to the AGM, Nuclear within 14 days, and approved at the subsequent, regularly scheduled meeting.

D6.5.2 -NOT USED-

D6.5.3 TECHNICAL REVIEW AND CONTROL

Activities which affect NUCLEAR SAFETY shall be conducted as follows:

a. Procedures, plans, manuals, and programs required by Specification D6.8 and other procedures, plans, manuals, and programs which affect NUCLEAR SAFETY, and changes thereto, shall be prepared, reviewed, and approved. Each such procedure, plan, manual, and program or change thereto shall be reviewed by an individual(s) other than the preparer, but who may be from the same organization as the preparer of the procedure, plan, manual, and program or change thereto. Programs, plans, manuals, and procedures other than plant administrative procedures will be approved as delineated by specific technical specifications, otherwise, as delineated in writing by the AGM, Nuclear, but not lower than the manager level. Such procedures, plans, manuals, and programs shall be reviewed periodically in accordance with administrative procedures.

   The AGM, Nuclear will approve plant administrative procedures, Security Plan Implementing Procedures, and Emergency Plan Implementing Procedures.

   Approval of temporary procedure changes which clearly do not change the intent of the approved procedure can be made by two members of the plant management staff, at least one of whom is a Certified Fuel Handler, as a minimum. The change shall be documented, reviewed, and approved by the designated approval authority for that procedure within 14 days of implementation.

b. Proposed changes or modifications to plant systems or equipment that affect NUCLEAR SAFETY shall be reviewed by an individual(s) other than the individual(s) who designed the modification, but who may be from the same organization as the individual(s) who designed the modifications. Such modifications shall be approved by the AGM, Nuclear or his designee as delineated in writing, but not lower than the manager level.

c. Proposed tests and experiments which affect NUCLEAR SAFETY and are not addressed in the USAR shall be reviewed by an individual(s) other than the individual(s) who prepared the proposed test or experiment. Such tests or experiments shall be approved by the AGM, Nuclear or his designee as delineated in writing, but not lower than the manager level.

d. Individuals responsible for reviews performed in accordance with Specification D6.5.3a, b, and c shall meet or exceed the qualification requirements of Section 4.4 of ANSI N18.1-1971. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such a review shall be performed by the review personnel of the appropriate discipline. A list of qualified reviewers for the independent reviews described in D6.5.3a, b, and c above shall be established by the AGM, Nuclear.

e. Events reportable pursuant to Specification D6.9.5 and violations of technical specifications shall be investigated and a report prepared which evaluates the event and which provides recommendations to prevent recurrence. Such reports shall be reviewed by the NSRAC and forwarded to the AGM, Nuclear.

D6.5.4 AUDITS

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

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

b. The performance and qualifications of the Rancho Seco facility technical staff at least once per year.

c. The result of actions taken to correct deficiencies occurring in facility equipment, structures, systems or methods of operation that affect NUCLEAR SAFETY at least once per 6 months for those changes not previously audited.

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

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

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

g. Any other area of facility activity considered appropriate by the AGM, Nuclear.

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

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

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

k. The REMP results at least once per year.

l. The ODCM and REMP Manual and implementing procedures thereof at least once per 2 years.

m. The PCP and implementing procedures for processing and packaging of radioactive wastes from liquid systems at least once per 2 years.

n. The performance of activities required by the Quality Assurance Program for Effluent Control and Environmental Monitoring at least once per year.

Audit reports of reviews encompassed by Specification D6.5.4 shall be forwarded to the AGM, Nuclear, the NSRAC Chairman, and the management positions responsible for the areas reviewed within 30 days after completion.

D6.6 LICENSEE EVENT REPORT ACTION

The following actions shall be taken for events which are reportable as LICENSEE EVENT REPORTS:

a. The Commission shall be notified and a report submitted per Specification D6.9.5 pursuant to the requirements of 10 CFR 20.405, 50.73, and 73.71, and

b. Each LICENSEE EVENT REPORT shall be reviewed by the NSRAC. Also, each report shall be reviewed and approved by the AGM, Nuclear, or designee.

D6.7 -NOT USED-

D6.8 PROCEDURES, PLANS, MANUALS, AND PROGRAMS

D6.8.1 Written procedures/plans/manuals/programs shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Safety Guide 33, November 1972.
b. The safe handling and storage of irradiated core components
c. Surveillance and test activities on equipment required for long term safe storage of irradiated core components

d. Security Plan implementation
e. Emergency Plan implementation
f. Fire Protection Program Plan implementation
g. PCP implementation
h. ODCM implementation
i. REMP MANUAL implementation
j. Quality Assurance for Effluent Control and Environmental Monitoring using the guidance of Regulatory Guide 4.15, Revision 1, February 1979
k. Certified Fuel Handler Training Programs implementation
l. Quality Assurance Program implementation

D6.8.2 Each procedure/plan/manual/program of Specification D6.8.1 above and changes thereto shall be reviewed and approved as set forth in Specification D6.5.

D6.8.3 The following programs shall be established, implemented, and maintained:

a. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBER(S) OF THE PUBLIC from radioactive effluents as low as reasonably achieveable. The program (1) shall be contained in the ODCM, (2) shall be implemented by Administrative, Chemistry, and Operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to 10 CFR Part 20, Appendix B, Table II, Column 2,

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.106 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to MEMBER(S) OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the OPERABILITY and use of the gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the OPERABILITY and use of the liquid effluent treatment system to ensure that the appropriate portions of this system is used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 8 1/3 percent (1/12) of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR 50.

8) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the SITE BOUNDARY conforming to the doses associated with 10 CFR 20, Appendix B, Table II, Column 1,

9) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual and quarterly doses to a MEMBER(S) OF THE PUBLIC from Tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

11) Limitations on the annual dose or dose commitment to MEMBER(S) OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

b. Radiological Environmental Monitoring Program (REMP)

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the REMP MANUAL, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the REMP manual,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

D6.8.4 Each program of Specification D6.8.3 above and changes thereto shall be reviewed and approved as set forth in Specification D6.5.

D6.9 REPORTING REQUIREMENTS

D6.9.1 ROUTINE REPORTS

In addition to the applicable reporting requirements of Title 10 to the Code of Federal Regulations the following reports shall be submitted to the NRC Regional Administrator of the Region V Office unless otherwise noted.

D6.9.2 ANNUAL RADIOLOGICAL REPORTS

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

D6.9.2.1 Annual Occupational Radiation Exposure Report

The Annual Occupational Radiation Exposure Report for the previous calendar year shall be submitted to the Commission within the first calendar quarter of each calendar year in compliance with 10 CFR 20.407.

D6.9.2.2 Annual Exposure Report

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

D6.9.2.3 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM, (2) the REMP MANUAL, and (3) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

D6.9.3 SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

The Semiannual Radioactive Effluent Release Report covering the operation of the unit during the previous 6 months shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR 50.

D6.9.4 ANNUAL REPORT

A routine report consisting of shutdown statistics, a narrative summary of shutdown experience and major maintenance of equipment required for long term safe storage of irradiated core components, and tabulations of facility changes, tests, or experiments required pursuant to 10 CFR 50.59(b) shall be submitted on an annual basis to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Office, postmarked no later than 30 days following the twelve month period covered by the report.

D6.9.5 LICENSEE EVENT REPORT

The types of events listed in 10 CFR 20.405, 50.73, and 73.71 shall be the subject of LICENSEE EVENT REPORTS, submitted to the U.S. Nuclear Regulatory Commission (NRC), Document Control Desk, Washington, D.C. 20555, within the time requirements of 10 CFR 50.73. An additional copy shall also be submitted to the Regional Administrator of the Region V Office. The written report shall include a completed copy of a LICENSEE EVENT REPORT form, pursuant to 10 CFR 50.73 and the guidance of NUREG-1022, and a description of corrective actions and measures that are designated to prevent recurrence. Supplemental reports may be required to fully describe final resolution of the occurrence. For corrected or supplemental reports, another LICENSEE EVENT REPORT shall be completed, and reference shall be made to the original report date, pursuant to the requirements of 10 CFR 20.405, 50.73, or 73.71 as appropriate.

D6.9.6 ENVIRONMENTAL REPORTS

a. When a change to the plant design or to plant activities is planned which would have a significant adverse effect on the environment or which involves an environmental matter or question not previously reviewed and evaluated by the NRC, a report on the change will be made to the NRC prior to implementation. The report will include a description and evaluation of the change, including a supporting benefit-cost analysis.

ADMINISTRATIVE CONTROLS

b. Changes or additions to permits and certificates required by Federal, State, local, and regional authorities for the protection of the environment will be reported. When the changes required by the concerned agency are submitted for approval, they will also be submitted to the NRC for information. The submittal will include an evaluation of the environmental impact of the change.

D6.9.7 SPECIAL REPORTS

Special reports shall be submitted to the Regional Administrator of the Region V Office within the time period specified for each report.

D6.10 RECORD RETENTION

D6.10.1 The following records shall be retained for at least 5 years:

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

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

c. LICENSEE EVENT REPORTS.

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

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed-source leak tests and results.

i. Records of annual physical inventory of all sealed-source material of record.

j. Records and logs of plant systems and equipment operation in the PERMANENTLY DEFUELED MODE.

D6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the USAR.

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

c. Records of facility radiation and contamination surveys.

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

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

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

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

h. Records of in-service inspections performed previously pursuant to the Appendix A Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

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

k. Records of meetings of the PRC and MSRC.

l. Records of the environmental qualification of safety-related electrical equipment previously generated pursuant to Appendix A Specification 6.14.

m. Records for the Radiological Environmental Monitoring Program.

n. Records of the service lives of all hydraulic and mechanical snubbers previously generated pursuant to Appendix A Specification 6.10.2n, including the date at which the service life commences and associated installation and maintenance records.

o. Records of reviews performed for changes made to the ODCM, REMP MANUAL, and the PCP.

p. Records of information important to the safe and effective decommissioning of the facility.

q. Records of meetings of the NSRAC.

D6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR 19 and 10 CFR 20, and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

D6.12 HIGH RADIATION AREA

D6.12.1 In lieu of the "control device" or "alarm signal" required by 10 CFR 20.203(c)(2).

a. Each High Radiation Area in which the intensity of radiation is equal to or greater than 100 mrem/hr but less than 1,000 mrem/hr shall be barricaded and conspicuously posted as a High Radiation Area, and entrance thereto shall be controlled by issuance of a Radiation Work Permit. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is equal to or greater than 1,000 mrem/hr shall be subject to the provisions of Specification D6.12.1a above, and, in addition, locked doors shall be provided to prevent unauthorized entry into such area. The keys shall be maintained under the administrative control of the on duty Shift Supervisor. In lieu of locked doors, certain areas within the Reactor Building may employ conspicuous visible or audible signals such that an individual is made aware of the presence of the High Radiation Area (>1,000 mrem/hr).

D6.13 PROCESS CONTROL PROGRAM (PCP)

D6.13.1 The required content of the PCP is defined in Specification D1.6.

D6.13.2 Licensee initiated changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification D6.10.2o. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

   2) A determination that the change will maintain the overall conformance of the processed waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the NSRAC and approval by the AGM, Nuclear.

D6.14 OFFSITE DOSE CALCULATION AND RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUALS

D6.14.1 The required content of the ODCM is defined in Specification D1.7.

D6.14.2 The required content of the REMP MANUAL is defined in D1.10.

D6.14.3 Licensee initiated changes to the ODCM or REMP MANUAL:

a. Shall be documented and records of reviews performed shall be retained as required by Specification D6.10.2o. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.106, 40 CFR 190, 10 CFR 50.36a, and Appendix I to 10 CFR 50 and not adversely impact the accuracy or reliability of effluent dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the NSRAC and approval by the AGM, Nuclear.

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM and/or REMP MANUAL as a part of or concurrent with the Semiannual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM and/or REMP MANUAL was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

Appendix B Technical Specification Pages Affected by

Proposed Amendment No. 182, Revision 2

Appendix B

Remove	Insert
i	i
ii	---
1 - 50	---

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

APPENDIX B

Appendix B has been deleted in its entirety.

ATTACHMENT II

Safety Analysis and

No Significant Hazards Consideration

for

PA-182, Rev. 2

Description of Change

Proposed Amendment No. 182, Revision 2 (PA-182, Rev. 2) creates a new appendix, Appendix C, to Facility Operating License No. DPR-54 and deletes Appendix B. Appendix C, the Permanently Defueled Technical Specifications (PDTS), will be followed in lieu of the existing Appendix A Technical Specifications for the current extended outage that began in June 1989. When the Facility Operating License is amended to a Possession-Only License (POL), the PDTS will become the POL Technical Specifications with some minor modifications. The PDTS define a new operating mode, titled Permanently Defueled Mode (PDM), which applies to Rancho Seco's current defueled condition. Movement of fuel into the Reactor Building, which is not allowed without prior NRC approval, would take the plant out of the PDM and the PDTS (Appendix C), and cause the plant to revert back to the Appendix A Technical Specification requirements.

The PDTS provide Limiting Conditions for Operation and Surveillance Requirements for the systems and components required to be Operable to ensure long term safe storage of irradiated core components in the Spent Fuel Pool (SFP). The PDTS also provide Administrative Control requirements for the overall safe operation of the facility. Implementation of the PDTS will ensure the continued protection of the health and safety of the public during the PDM.

Reason For Change

The District developed the proposed PDTS based on (1) Rancho Seco's current defueled condition, (2) the minimal potential consequences associated with the accidents considered credible in the defueled condition, and (3) the District's acknowledged intent to not operate Rancho Seco as a nuclear facility in the future. The fuel is removed from the reactor vessel and is now stored in the SFP. Plant systems are being layed up in either a wet or dry condition, as appropriate, to limit degradation during the current extended outage. The PDTS provide (1) the level of plant protection needed to deal with the credible accidents postulated in the defueled condition, (2) long term safe storage of irradiated nuclear fuel and control components, and (3) the continued protection of public health and safety.

The PDTS provide a clear and concise set of Specifications applicable in an extended outage and defueled condition. The PDTS provide relief from those Appendix A Technical Specification requirements which do not apply or are not necessary to protect the health and safety of the public in the PDM.

The PDTS are submitted as an addition to the current Facility Operating License and are intended to be Appendix C to Facility Operating License No. DPR-54. Upon NRC approval, the PDTS become applicable in lieu of the Appendix A Technical Specifications while the plant is in the PDM. Movement of the fuel into the Reactor Building takes the plant out of the PDM and causes the plant to revert back to the Appendix A Technical Specification requirements.

Evaluation and Basis For Safety Findings

The primary difference between the proposed PDTS and the current Rancho Seco Technical Specifications is the range of postulated accidents that these two sets of Specifications are required to address, and the range of potential radiological consequences resulting from abnormal plant conditions in the various applicable modes. Based on a review of these conditions and credible accidents, the PDTS were developed to provide the necessary level of protection.

The Updated Safety Analysis Report (USAR), Chapter 14, contains two accidents or conditions that are considered credible in the PDM:

1. Fuel Handling Accident

2. Complete Loss of All Unit a-c Power (LOOP)

These two accidents or conditions apply in the defueled condition and are considered in the analysis below.

Fuel Handling Accident

The spent fuel at Rancho Seco has undergone substantial decay since the reactor was shut down on June 7, 1989, thus the source term for a credible accident in the PDM is significantly less than that assumed for an operating plant. Essentially no radioactive iodine is present in the plant that could have a significant dose impact on members of the public, thus the calculated thyroid exposure levels resulting from credible accidents in the PDM are negligible. The primary radioisotope of concern from an offsite exposure (dose) standpoint following the dropped fuel assembly accident, the maximum credible accident in the PDM, is Krypton-85. Results of the analysis for the dropped fuel assembly accident in the current defueled condition show that the 2-hour integrated total body dose attributed to the maximum exposed individual is 0.013 rem. This calculated dose is an extremely small fraction of the 10 CFR 100 accident dose limit (25 rem), and is significantly less than the annual dose limit of 10 CFR 20 (0.5 rem) or the plume exposure Protective Action Guidelines. The District expects only limited fuel handling activities while the plant is in the PDM.

LOOP

During normal plant operations and post accident conditions it is imperative that electrical power be available to support equipment needed to operate the plant and mitigate the consequences of an accident. In the PDM, a LOOP would result in the loss of the Spent Fuel Pool Cooling (SFC) system. However, there is adequate time available (see the SFP decay heat load evaluation below) to take corrective action without a safety consequence in the event of a LOOP. It takes a minimum of four days following the loss of SFC before the SFP could begin to boil. Rancho Seco has six offsite power transmission lines, and has the capability to receive power directly from either the District's or PG&E's hydroelectric units in less than eight hours. An evaluation of the offsite electrical grid for Rancho Seco, performed pursuant to 10 CFR 50.63, Station Blackout, verified the stability of the Western grid. The probability of a LOOP at Rancho Seco, as evaluated in accordance with the guidelines of Regulatory Guide 1.155, is less than once per 20 years. Therefore, the emergency diesel generators are not required to ensure power availability to support SFC equipment in the event of a LOOP. An alternate power supply can be made available well within the minimum time required to take corrective action to restore SFC. A LOOP will not result in the release of a significant amount of SFP inventory or radioactivity.

SFP Decay Heat Load

The controls required to protect the spent fuel in the PDM are predicated primarily on the level of decay heat in the SFP. The decay heat load for the SFP in the defueled condition was calculated using the methodology described in ANSI/ANS 5.1-1979 and BTP ASB 9-2. The decay heat load in the SFP, as a function of time, is shown in Table 1 (see page 56 of this Safety Analysis Report (SAR)). This table provides a conservative estimate of the SFP heatup rate following a loss of SFC. Specifically, Table 1 provides, as a function of calendar date, the time required for the SFP to reach 212°F from an initial temperature of 120°F, and the time required to boil 6.75 feet of water from the SFP following a loss of SFC. The 6.75 feet was chosen since the SFP water level is normally maintained above 37 feet (the low-low level alarm) and, as conservatively evaluated previously, must be maintained at 30 feet 3 inches to limit the dose rate at the SFP surface to 2.5 mr/hr when 1080 spent fuel assemblies that have decayed 3 days following irradiation at 100% power is stored in the SFP. This SFP surface dose rate calculation is very conservative since the reactor was not operated at 100% power as assumed in the dose rate calculation, the number of spent fuel assemblies stored in the SFP is much less than 1080, and the actual decay time of the 493 spent fuel assemblies in the SFP is significantly higher than the 3 days assumed. No safety implication exists for a significantly lower SFP level as long as personnel exposure is monitored and maintained as low as is reasonably achievable and an appropriate level of cooling is maintained in the SFP. PDTS Specification D3/4.1 provides limits for the SFP surface dose rate.

The results presented in Table 1 are conservative since they are based on the following conservative assumptions:

1. The power history for each reactor run is calculated to have been accumulated as the effective full power days (EFPD) of the run at 100% power at the end of the run (i.e., the 245 EFPD run ending on June 7, 1989 is assumed to have occurred at 100% power in the last 245 days before June 7, 1989, when in fact it was spread out over a 48 month period).

2. The decay heat load calculations include a conservative factor of 10%.

3. There is no heat loss from the SFP due to evaporation or ambient losses, except after the SFP reaches the boiling point.

As seen in Table 1, upon a loss of SFC the minimum time required to reach boiling in the SFP is 96.12 hours as of June 7, 1990, and increases to 143.85 hours on June 7, 1991. Thus, a minimum of 4 days, and in the near future 6 days, is available to restore SFC prior to the occurrence of boiling following a loss of SFC. In addition, if the extra time available through boil-off of 6.75 feet of the 37 foot normal minimum water depth is taken into account, a minimum of 11.86 days as of June 7, 1990, and 17.74 days as of June 7, 1991, is available for corrective actions to be implemented to restore SFC. If boil-off of SFP water were to occur due to a loss of SFC, a simple addition of water to the SFP would extend the time to implement corrective actions to restore SFC, if necessary. The safety analysis for Specification D3/4.2 (see pages 15 and 16 of this SAR) lists several options available for providing makeup water to the SFP.

No degradation effects on the fuel and cladding is associated with SFP boiling since the fuel is designed to operate at significantly higher coolant temperatures than 212°F. The effect of thermal stresses on the structural support of the SFP have been analyzed and found acceptable up to a SFP temperature of 212°F. Therefore, a 212°F SFP temperature does not impose a safety hazard.

Criticality Control

Criticality control in the SFP is achieved through the use of high density fuel storage racks that contain a neutron absorbing material (Boraflex) within the rack plates. These racks are designed to hold 1,080 fuel assemblies at 4.0 weight percent enrichment with unborated water in the SFP while still maintaining K_{eff} less than 0.95. The current fuel inventory in the SFP is 493 assemblies with a maximum enrichment prior to burnup of 3.21 weight percent. District calculations show that the estimated maximum synthesized enrichment (remaining U-235 plus fissile Pu) for any assembly in the SFP is 2.573 weight percent, thus providing an even greater shutdown margin. Also, control assemblies are stored in several spent fuel assemblies. This provides additional neutron absorption material; therefore, no boron is required in the SFP to maintain an adequate shutdown margin.

The SFP storage rack neutron absorption capability is monitored through a coupon sampling program. Boraflex material samples are periodically removed from the SFP for testing to verify rack performance. To date, the removed coupons have provided verification of the Boraflex integrity, absorption capability, and spent fuel storage rack performance as designed. In addition, the PDTS contain new requirements to monitor the SFP water for chloride and fluoride levels to assure rack and fuel structural integrity is maintained over time. The combination of these two programs provides assurance that the spent fuel storage racks will remain structurally sound and perform their intended criticality control function.

An additional event considered for the PDM is the drop of a spent fuel cask. This event is addressed in USAR sections 9.8.2.4 and 9.8.2.5. Because of the many mechanical and electrical interlocks and administrative controls associated with the Turbine Building Gantry Crane, which are designed to prevent movement of a cask over spent fuel assemblies, the drop of a cask onto spent fuel assemblies is not considered a credible event. In addition, plant configuration and crane interlocks prevent a spent fuel cask from being lifted to a height more than 26 feet in air or 33 feet 6 inches in water during the cask transport process. Casks are required to be rated for a 30 foot drop in air pursuant to 10 CFR 71. This height in air is equivalent to more than a 40 foot drop in water. Therefore, cask integrity is assured in the unlikely event a cask is dropped.

In addition to the above evaluation of the accidents and conditions considered credible in the PDM, several Generic Letters, as listed below, are considered in the preparation of the PDTS as follows:

Generic Letter 89-01, IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR THE PROCESS CONTROL PROGRAM

b. Generic Letter 89-14, LINE-ITEM IMPROVEMENTS IN TECHNICAL SPECIFICATIONS - REMOVAL OF THE 3.25 LIMIT ON EXTENDING SURVEILLANCE INTERVALS

c. The previously docketed submittals made pursuant to Generic Letter 88-12, REMOVAL OF FIRE PROTECTION REQUIREMENTS FROM TECHNICAL SPECIFICATIONS, are acknowledged in this submittal by omission of the appropriate Technical Specifications. Amendment No. 115 to Facility Operating License DPR-54 relocated fire protection related Technical Specifications requirements to the Fire Protection Plan pursuant to Generic Letter 88-12.

The following is a line by line discussion of the Appendix A Technical Specifications and their disposition in relationship to the PDTS.

Technical Specification Number	Discussion
1.1	The term Rated Power is not applicable in the PDM and is not included in the PDTS.
1.2 (D1.1, D1.2)	Definitions 1.2.1 through 1.2.12 and 1.2.14, and Table 1.2-1 are related to plant operations or non defueled activities, are not applicable in the defueled condition, and are not included in the PDTS. Specification 1.2.13 (Action) is retained in the PDTS as Specification D1.1. Also, a definition for the mode of applicability for the PDTS (the PDM) is included in the PDTS as Specification D1.2.
1.3 (D1.3)	The definition for the term Operable is modified to remove the requirement for an operable emergency power source while the plant is in the PDM. The current Technical Specifications do not require the emergency power supplies to be Operable when the plant is in a mode below Heatup-Cooldown. Also, the part of the existing definition that addresses the requirement for an Operable redundant system is not applicable in the PDM. A revised definition of Operable is included in the PDTS as Specification D1.3.
1.4 (D1.4.1)	The definitions for Protection Instrumentation Logic apply to systems such as SFAS and RPS that are not in service in the PDM, except for Specification 1.4.1, Instrument Channel. The remaining terms are not used in the PDTS and are not included in the Definition section. The definition for Instrument Channel is retained in the PDTS as Specification D1.4.1.
1.5 (D1.4.2, D1.4.3, D1.4.4, D1.4.5)	The definitions from this Specification that are used in the PDTS, i.e. Channel Test, Instrument Channel Check, and Instrument Channel Calibration, are included in the PDTS as Definitions D1.4.2, D1.4.3, and D1.4.4, respectively. The remaining terms are not used in the PDTS and are not included in the Definition section. A new definition, Functional Test, is added to clarify the test currently performed on the Fuel Storage Building fuel handling bridge interlocks. This definition is included in the PDTS as Specification D1.4.5.
1.6	These definitions describe core reactivity conditions that are not applicable when the plant is in the PDM; therefore, these definitions are not used and are not included in the PDTS.
1.7	This Specification defines Containment Integrity which is not applicable in the PDM. Therefore, this definition is not included in the PDTS.

Technical Specification Number	Discussion
1.8 (D1.9)	The description for Licensee Event Report (LER) is included in the PDTS as specification D6.9.5. A reference to this LER Specification is retained in the PDTS Definition section as Specification D1.9.
1.9 (Table D1-1, D4.0.2)	The definitions and notation for the surveillance intervals applicable in the PDTS are included in the PDTS as Specification Table D1-1. The requirement to not exceed 3.25 surveillance time intervals for any three consecutive intervals is revised in accordance with Generic Letter 89-14. This requirement is relocated and included in the PDTS as Specification D4.0.2 in accordance with NRC comments, the Standard Technical Specifications (STS), and Generic Letter 87-09.
1.10 (D1.11)	The term Safety is changed to Nuclear Safety, and its definition is editorially modified and included in the PDTS as Specification D1.11.
1.11	Specification 1.11 was previously relocated to the Fire Protection Plan in accordance with Generic Letter 88-12 and License Amendment No. 115, and is not included in the PDTS.
1.12	This definition, Staggered Test Basis, is not used and thus is not included in the PDTS.
1.13 (D1.6)	The definition for the Process Control Program (PCP) is included in the PDTS as Definition D1.6 and is modified in accordance with Generic Letter 89-01. The PCP is provided in Attachment VI for NRC use as a reference. The PCP in Attachment VI is the same as the one included in the PA-182, Rev. 0 submittal package. The PCP is included in the PA-182, Rev. 2 submittal package for completeness.
1.14	The term Solidification is replaced by a more generic term, Processing, and its definition is relocated to the PCP per the guidelines of Generic Letter 89-01. The term Processing allows the use of new radwaste processing techniques that meet NRC and State requirements in addition to Solidification. The PCP is provided in Attachment VI in accordance with Generic Letter 89-01 for NRC use as a reference. The PCP in Attachment VI is the same as the one included in the PA-182, Rev. 0 submittal package. The PCP is included in the PA-182, Rev. 2 submittal package for completeness.

Technical Specification Number	Discussion
1.15 (D1.7)	The definition for the Offsite Dose Calculation Manual (ODCM) is included in the PDTS as Specification D1.7 and is modified in accordance with Generic Letter 89-01. The ODCM is provided in Attachment IV for NRC use as a reference. The ODCM in Attachment IV is the same as the one included in the PA-182, Rev. 1 submittal package. The ODCM is included in the PA-182, Rev. 2 submittal package for completeness.
1.16 (D1.12, D1.13)	The term Restricted Area is changed to Industrial Area and is included in the PDTS as Specification D1.12. The definition for Industrial Area is worded in accordance with NRC comments. The term Unrestricted Area is used in the PDTS, and a definition for this term is included in the PDTS as Specification D1.13. The definition for Unrestricted Area is a slightly modified version of the definition contained in the Standard Radiological Effluent Technical Specifications (NUREG-0472). Because licensed activities are conducted within the Industrial Area, and credible accidents in the PDM do not cause the plume exposure Protective Action Guidelines to be exceeded at the Industrial Area boundary, the unrestricted area for radiation protection purposes is at or beyond the Industrial Area boundary and not the Site Boundary.
1.17 (D1.5)	The term Site Boundary is used in the PDTS and is included in the PDTS as Specification D1.5. The definition for Site Boundary is obtained from the STS (NUREG-0103).
1.18	The definition for Dose Equivalent I-131 is relocated to the ODCM per Generic Letter 89-01.
1.19 (D1.8)	The definition for Member(s) Of The Public is retained and included in the PDTS as Specification D1.8.
1.20	The definition for Dewatering has been replaced by a more generic term, Processing, and relocated to the PCP per Generic Letter 89-01. The term Processing allows the use of new radwaste processing techniques that meet NRC and State requirements in addition to Dewatering.
1.21	This definition, Maximum Exposed (Hypothetical) Individual, is relocated to the ODCM per Generic Letter 89-01.

Technical Specification Number	Discussion
1.22 (D1.10)	The definition for the Radiological Environmental Monitoring Program (REMP) Manual has been modified per Generic Letter 89-01 and is included in the PDTS as Specification D1.10. The REMP Manual is provided in Attachment VIII in accordance with Generic Letter 89-01 for NRC use as a reference. The REMP Manual provided in Attachment VIII is an editorially modified version of the one provided previously in the PA-182, Rev. 0 submittal package. The REMP Manual included in the PA-182, Rev. 2 submittal package replaces the REMP Manual submitted previously.
1.23	This definition, Liquid Effluent Radwaste Treatment System, is relocated to the ODCM per Generic Letter 89-01.
1.24	The term Ventilation Exhaust Treatment Systems is relocated to the ODCM per Generic Letter 89-01.
1.25	This definition, Purge - Purging, is relocated to the ODCM per Generic Letter 89-01.
1.26	This definition, Venting, is relocated to the ODCM per Generic Letter 89-01.
1.27	This definition, E-BAR, is for radionuclide concentrations in the Reactor Coolant System (RCS) based on beta and gamma activity, is not applicable in the PDM, and is not included in the PDTS.
2	Section 2 of the Appendix A Technical Specifications provides limits for reactor and RCS operation. This section is designed to maintain the integrity of the fuel cladding during reactor operation and prevent fission product release to the RCS. These Specifications are not applicable in the PDM. Therefore, this section is not included in the PDTS.
3.0.1 (D3.0.1)	This Specification has been modified to reflect the single mode of applicability of the PDTS (the PDM), and is included in the PDTS as Specification D3.0.1.
3.0.2 (D3.0.2)	This Specification is retained in the PDTS with only a slight editorial change, and is included in the PDTS as Specification D3.02.
3.0.3	This Specification provides the time limit within which a mode change must be made when an LCO is not met. This Specification is not applicable when the PDTS are applicable since the plant can only be in one mode. Therefore, this Specification is not included in the PDTS.

Technical Specification Number	Discussion
3.0.4	This Specification provides restrictions for entry into an operational mode. This Specification is not applicable when the PDTS are applicable since the plant can only be in one mode. Therefore, this Specification is not included in the PDTS.
3.1 (D3/4.5)	Specifications 3.1.1.1 through 3.1.9.2 are not required to be met below the Heatup-Cooldown mode and are not included in the PDTS. These Specifications are not applicable since they provide protection of the RCS and the reactor core during reactor operation modes. These requirements provide no additional protection in the PDM. A Specification addressing SFP water chemistry requirements is adapted from the reactor coolant water chemistry requirements of existing Specification 3.1.5, and is included in the PDTS as Specification D3/4.5. This new requirement provides appropriate SFP water chemistry limits to ensure the integrity of the fuel assemblies, the SFP liner, and the fuel storage racks is protected against potential degradation in the PDM. Fifteen years of plant operating experience proves these limits are adequate to protect SFP components from degradation.
3.2	Specification 3.2.1 provides for adequate boration control of the RCS and ensures the ability to bring the reactor to Cold Shutdown following an accident. Since the heat and criticality source (the core) has been removed, the need to provide this type of protection does not exist. The requirements of Specification 3.2.1 are not included in the PDTS. During power plant operations, Technical Specification 3.2.2 assures, in cold conditions, that the core will be covered with water by preserving the physical integrity of the reactor vessel, thereby protecting the health and safety of the public. This is accomplished by specifying requirements that are designed to protect the vessel from an inadvertent pressurization when the plant is in a cold operating condition, i.e., when the RCS is at or below 350° F with fuel in the reactor and the RCS not open to the Reactor Building atmosphere. With the reactor vessel defueled and RCS operation no longer necessary or desired, overpressure protection, to the extent defined in Specification 3.2.2, is not required or needed to mitigate the consequences of a design basis accident considered credible in the PDM. In the PDM, overpressure protection is needed only to help protect and preserve plant equipment for possible future use, not to ensure the health and safety of the public. Because overpressure protection for the RCS is not required to ensure Nuclear Safety in the PDM, overpressure protection requirements are not included in the PDTS. The

Technical Specification Number	Discussion
3.2 (Cont.)	necessary level of overpressure protection in the PDM is considered and addressed in the layup plan for the RCS. The NRC granted an interim waiver from compliance with Technical Specification 3.2.2 in NRC to SMUD letter dated May 14, 1990. The interim waiver will remain in effect until the PDTS are approved.
3.3	Specifications 3.3.1 through 3.3.2 address requirements for the Emergency Core Cooling, Reactor Building Emergency Cooling, and Reactor Building Spray systems and are not applicable below the Heatup-Cooldown mode. Since the reactor is defueled, there are no postulated accidents that require these systems to be operational. Therefore, the requirements of these Specifications are not included in the PDTS.
3.4	Specifications 3.4.1 through 3.4.2 specify the components required to be Operable to maintain the steam generators in service as an RCS heat removal source. These Specifications are not applicable below the Heatup-Cooldown mode since below this mode the RCS is not at a temperature where the steam generators are required to be in service. Therefore, these Specifications are not included in the PDTS.
3.5.1	Specifications 3.5.1.1 through 3.5.1.11 require instrumentation associated with the Reactor Protection System (RPS), Safety Features Actuation System (SFAS), Emergency Feedwater Initiation And Control System (EFIC), and other reactor operations safety equipment to be Operable for startup of the plant. These Specifications are not applicable below the Heatup-Cooldown mode. Because the reactor is defueled and no postulated accidents of any significant consequence are postulated to occur in the Reactor Building during the PDM, these Specifications are not included in the PDTS.
3.5.2	Specifications 3.5.2.1 and 3.5.2.2 provide limits on core power distribution and control rod Operability and are not applicable during any of the shutdown modes. Since the reactor is defueled, these Specifications are not applicable in the PDM and are not included in the PDTS.
3.5.3	Specification 3.5.3 provides setpoints for SFAS and is not applicable in the Cold Shutdown/Refueling Shutdown/Refueling Operations modes. Since the accidents mitigated by SFAS actuation are not credible in the defueled condition, this Specification is not included in the PDTS.

Technical Specification Number	Discussion
3.5.4	Specifications 3.5.4.1 and 3.5.4.2 provide core axial imbalance and radial tilt limits for reactor operation. These Specifications are not applicable during the Cold Shutdown/Refueling Shutdown/Refueling Operations modes and are not included in the PDTS since there is no fuel in the core.
3.5.5	Specification 3.5.5 provides a list of accident monitoring equipment that must be Operable for reactor operations. As stated in Note (1) of Technical Specification Table 3.5.5-1, this Specification is not applicable in the Cold Shutdown/Refueling Shutdown/Refueling Operations modes and is therefore not included in the PDTS. The postulated accidents these instruments are required to monitor are not credible in the PDM.
3.5.6	Specification 3.5.6 provides setpoints for the EFIC system. The EFIC system is not required to be Operable in the Cold Shutdown/Refueling Shutdown/Refueling Operations modes. This Specification is not included in the PDTS since the steam generators that the EFIC system supports are not required in the PDM.
3.5.7	Specification 3.5.7 provides a list of instrumentation required to achieve and maintain Hot Shutdown from outside the Control Room. This capability is required when the plant is in Hot Shutdown or above and is not required in the Cold Shutdown/Refueling Shutdown/Refueling Operations modes. Therefore, this Specification is not included in the PDTS.
3.6	Specifications 3.6.1 through 3.6.8 provide containment integrity requirements to ensure that radiological releases to the environment following an accident inside the Reactor Building are maintained within 10 CFR 100 limits. Since there are no postulated accidents that could occur inside containment in the PDM whose consequences could come anywhere near the 10 CFR 100 limits, these Specifications are not included in the PDTS.
3.7	Specifications 3.7.1 through 3.7.4 provide Operability requirements for the normal and emergency power sources which must be met before the reactor can be brought critical. As previously addressed in this submittal (see pages 3 and 4 of this Safety Analysis Report (SAR)), a significant amount of time is available to take corrective action to restore offsite power in the unlikely event of a LOOP. Therefore, an emergency power supply is not required, and these Specifications are not needed to ensure power is provided in a timely manner to support equipment in the PDM. These Specifications are not included in the PDTS.

Technical Specification Number	Discussion
3.8 (D3/4.3, D3/4.4)	Specification 3.8 provides requirements for fuel handling operations in the Spent Fuel Building and the Reactor Building. The Specification 3.8 requirements associated with the Spent Fuel Building are included in the PDTS. Specification 3.8.1, which addresses the radiation monitoring of the spent fuel storage area, was previously evaluated and approved for use at Rancho Seco by the NRC and is retained in the PDTS as Specification D3/4.4. Several enhancements are proposed to the existing spent fuel storage area radiation monitor requirements as follows:

1. The applicability of the proposed Specification is increased to "whenever irradiated fuel assemblies are in the spent fuel pool" instead of just during fuel handling operations to reflect the importance of monitoring the spent fuel storage area in the PDM.

2. A reporting requirement for the inoperability of a spent fuel storage area monitor for more than 7 days is added to the proposed Specification. This additional requirement ensures timely action is taken if fixed monitoring equipment is inoperable, thereby further ensuring the protection of individuals required to perform duties in the spent fuel storage area.

3. The instrument channel check surveillance requirement frequency is increased from weekly to every 12 hours during fuel handling operations to ensure the monitor is available to protect individuals in the spent fuel storage area.

When fixed radiation monitoring equipment is inoperable, portable radiation survey instrumentation that has the appropriate ranges and sensitivity is used to protect individuals in the spent fuel storage area until fixed radiation monitoring equipment is returned to operable status. This use of portable instrumentation when fixed instrumentation is inoperable was previously evaluated by the NRC, found acceptable, and included as part of the original licensing basis for Rancho Seco. Also, the proposed surveillance requirements, one of which is more conservative than the existing requirements, were previously reviewed and approved by the NRC. Several years of operating experience has proven that these requirements are adequate to protect the safety of plant personnel and the health and safety of the public. The proposed Specification is consistent with the requirements contained in the STS (NUREG-0103) for the spent fuel storage area radiation monitoring instrumentation.

Technical Specification Number	Discussion
3.8 (D3/4.3, D3/4.4) (Cont.)	The use of radiation monitoring equipment in the spent fuel storage area ensures that early notification of excessively high radiation is provided to protect individuals who perform duties in the spent fuel storage area. The area radiation monitors addressed in Specification 3.8.1 that are associated with the Reactor Building are not included in the PDTS since no fuel handling operations are allowed in the Reactor Building while the plant is in the PDM. The radiation monitors associated with the Reactor Building are not included in the PDTS since no fuel handling in the reactor building is allowed in the PDM.

Specifications 3.8.2 through 3.8.11 and 3.8.13 are associated · fuel movement in the Reactor Building, are not applicable in PDM, and are not included in the PDTS.

Specification 3.8.12 restricts the movement of loads over sp... fuel, except for fuel assemblies and control components, and .s retained in the PDTS as Specification D3/4.3. This limitation is consistent with the assumptions made in the accident analysis for a dropped fuel assembly, and assures in the event a fuel assembly is dropped that (1) the activity released would be limited to the activity contained in a single fuel assembly, and (2) any possible distortion of fuel in the SFP will not result in the formation of a critical array. Retaining the existing licensing basis for Rancho Seco in the PDTS provides and maintains an acceptable level of safety in the PDM.

The handling of loads in the fuel storage building has been previously evaluated by the NRC and is described and analyzed in USAR Section 9.8.2.5. This heavy load Specification meets the guidelines established in NUREG-0612 for the control of heavy loads at nuclear power plants and provides reasonable assurance that no heavy loads will be handled over irradiated fuel. As a result, the potential for accidental load drops over spent fuel is minimized. Several years of operating experience has shown that these requirements are sufficient to protect the safety of plant personnel and the health and safety of the public.

Technical Specification Number	Discussion
3.9 (D3/4.1, D3/4.2)	Specification 3.9.1 is retained in the PDTS as Specification D3/4.2 and editorially rewritten to clarify the requirements for maintaining SFP temperature in the PDM.

The SFP cooling system normally maintains the SFP below 100°F. Historically, inoperability of the SFP cooling system concurrent with elevated pool temperatures (≥140°F) has not occurred at Rancho Seco. Limiting the SFP bulk coolant temperature to 180°F achieves the design objective of preventing spent fuel pool boiling and ensures personnel safety and public health and safety. The requirements specified in Specification D3/4.2 have been previously evaluated and approved by the NRC in Amendment No. 84 to Rancho Seco Operating License No. DPR-54 as adequate to prevent SFP boiling. Retaining the licensing basis that existed for Rancho Seco as an operating nuclear power plant provides and maintains an acceptable level of safety in the PDM.

If the SFP cooling system becomes inoperable or fails to maintain the SFP temperature below 140°F, one train of the Decay Heat Removal System (DHRS) is available for service to ensure the SFP bulk coolant temperature will not reach the boiling point. The intent of maintaining one train of DHRS available as an alternate means of cooling is to supplement or supplant, as necessary, the SFP cooling system until the SFP cooling system is repaired.

Use of one train of DHRS as a backup to the SFP cooling system has been previously evaluated and approved for use at Rancho Seco by the NRC in Amendment No. 84 to Operating License No. DPR-54, and determined adequate to prevent SFP boiling. The 'A' train of the DHRS is administratively maintained available and can be lined up to provide SFP cooling per the appropriate operating procedures.

In addition to an alternate means of cooling, several methods are available to provide makeup water to the SFP, if necessary. Inventory losses from the SFP are made up in accordance with the appropriate operating procedures to ensure a sufficient inventory of water is maintained. The options currently available for providing makeup water to the SFP are summarized below:

1. Water from the Miscellaneous Water Holdup Tank is pumped through a valve and pipe directly to the SFP.

2. Water is pumped from the Condensate Storage Tank to the Demineralized Water Header then to the SFP through a hose.

Technical Specification Number	Discussion
3.9 (D3/4.1, D3/4.2) (Cont.)	3. Water from the Folsom South Canal or Rancho Seco lake is pumped to the SFP via the diesel fire pump or the electric fire pump and a fire hose station.
	4. Service water from the pressurized Service Water System is supplied from the Turbine Deck to the SFP through a hose.

As plant closure activities progress, the SFP makeup water options may change via the 10 CFR 50.59 process through changes to the appropriate operating procedures.

Specifications 3.9.2 and 3.9.3 provide restrictions on the use of the DHRS as a SFP cooling source when DHR is required for RCS protection. In the PDM, the DHRS is not required for protection of the reactor. Therefore, limitations on the use of DHRS as a SFP cooling source are unnecessary and these Specifications are not included in the PDTS.

Specification 3.9.4, requires the reactor to shutdown if the SFP bulk coolant temperature reaches 180°F. This specification is not applicable in the PDM and is not included in the PDTS.

Specification 3.9.5, SFP Water Level, is retained in the PDTS as Specification D3/4.1 and has been enhanced to include an additional surveillance requirement that addresses monitoring the surface dose and temperature of the SFP when the SFP level is less than 37 feet. Also, the requirements are clarified to ensure the appropriate portions of the Specification are applied when fuel handling operations are and are not in progress, thereby optimizing the safety features of this Specification.

During fuel handling operations, a minimum of 37 feet of water shall be maintained in the SFP. This requirement is consistent with the existing requirement that was previously evaluated and approved for use at Rancho Seco by the NRC in Amendment No. 97 to Operating License No. DPR-54. Also, this requirement is a condition previously analyzed and described in the USAR (Section 9.8.2.5). Requiring 37 feet of water in the SFP during fuel handling operations ensures a minimum of 9 feet of water shielding over active fuel; thereby protecting fuel handling personnel during the movement of fuel assemblies. Retaining the existing licensing basis for Rancho Seco in the PDTS provides and maintains an acceptable level of safety in the PDM.

Technical Specification Number	Discussion
3.9 (D3/4.1, D3/4.2) (Cont.)	The SFP level may be less than 37 feet only when fuel handling operations are not in progress, the dose rate at the pool surface is $\leq$2.5 mRem/hr, and the pool bulk coolant temperature meets the requirements of Specification D3/4.2. The existing requirements for SFP level, which were previously evaluated by the NRC (Amendment No. 97), allow for a SFP level below 37 feet but are not as restrictive as the requirements proposed in PDTS Specification D3/4.1. The proposed Specification provides additional restrictions when the SFP level is below 37 feet than is provided in the existing licensing basis for the plant. Providing appropriate restrictions on allowing a SFP level below 37 feet ensures that adequate shielding, cooling, plant personnel safety, and protection of public health and safety are maintained when no fuel handling operations are in progress. A lower SFP level would accommodate any unanticipated maintenance or repair to the SFP while spent fuel is stored in the SFP. The SFP temperature and surface dose rate limits have been previously evaluated by the NRC (Amendments 84 and 97, respectively), and are conditions previously analyzed in the USAR (Sections 9.6.1 and 9.8.2.5, respectively). The basis for maintaining the SFP water level $\geq$37 feet at all times in the Appendix A Technical Specifications was to minimize the consequences of an iodine release following the design basis Fuel Handling Accident (FHA). Because the reactor has been shut down since June 1989, iodine is no longer a concern for a FHA and the need to maintain the SFP level $\geq$37 feet during times other than fuel handling operations does not exist. The PDM FHA analysis does not rely on a water partition factor to mitigate the consequences of this design basis accident. Kr-85 is the predominant isotope in the accident analysis for the FHA. Kr-85 will evolve from the SFP following a rupture of fuel rods regardless of the water level in the pool. In addition, the SFP configuration is such that in the event of a SFP piping failure, the water in the SFP will not drain down lower than 10 feet above active fuel. The NRC has previously evaluated and approved the allowance of a SFP water level less than 37 feet, and this condition is analyzed and described in the USAR.
3.10	Specification 3.10 provides a limit for the amount of Iodine-131 in the secondary side of the steam generators. Since no residual Iodine-131 exists in the secondary plant from previous reactor operation and there is no production mechanism for Iodine-131 in the PDM, this Specification is not included in the PDTS.

Technical Specification Number	Discussion
3.11	Specification 3.11 provides limits on the use of the Reactor Building Polar Crane and Auxiliary Hoist. Since there is no fuel in the reactor vessel, the safety consequences of a dropped load in the Reactor Building are negligible. This Specification is not required to help ensure the protection of the public's health and safety; therefore, this Specification is not included in the PDTS.
3.12	Specification 3.12, Snubbers, provides testing requirements for snubbers on safety-related systems when the plant is above Cold Shutdown, and on systems required for Nuclear Safety otherwise. No snubbers remain that are on systems required to be Operable according to the PDTS. Therefore, since no snubber is a required auxiliary of a system required to be Operable in the PDM, the requirements of Specification 3.12 are not included in the PDTS.
3.13	Specifications 3.13.1, 3.13.3 and 3.13.4 provide requirements for the Control Room/Technical Support Center (CR/TSC) and Reactor Building Purge Exhaust filter systems when containment integrity is required and reactor operation is intended. Containment integrity is not required and reactor operation is not allowed in the PDM. Maintaining CR/TSC Emergency Filtering System Technical Specifications is not required in the PDM since the source term is low enough, due to fission gas decay, that a fuel handling accident will not result in exposure to Control Room personnel anywhere near allowable habitability levels. Chlorine detectors are part of the CR/TSC Emergency Filtering System. The chlorine detector Operability requirements are included within the Operability requirements for the CR/TSC Emergency Filtering System. According to Specification 3.13, the CR/TSC Emergency Filtering System is only required to be Operable when containment integrity is required. Containment integrity is not required in the PDM; therefore, the CR/TSC Emergency Filtering System Operability requirements, and thus chlorine detector Operability requirements, are not applicable and are not included in the PDTS. Since chlorine detector operability is not required in the PDM, the existing surveillance requirements associated with chlorine detectors (Specification 4.10.1E.2 and Item 44c of Table 4.1-1) are also not applicable and are not included in the PDTS.

Technical Specification Number	Discussion
3.13 (Cont.)	Based on previous NRC evaluations (Amendment Nos. 70 and 97 to Rancho Seco Operating License No. DPR-54), a chlorine detector Specification is not required when the plant is shutdown. Retaining the licensing basis that existed for Rancho Seco as an operating nuclear power plant provides and maintains an acceptable level of safety for Rancho Seco in the PDM. No accidents considered credible in the PDM require operator action to prevent the consequences of a radioactive release from exceeding 10 CFR 100 limits. Since the plant is shutdown and defueled and the consequences of a credible accident in the PDM would expose Control Room personnel to much less than 5 rem whole body, General Design Criterion 19 of 10 CFR 50, Appendix A is met without including a requirement for chlorine detectors in the PDTS. Also, as of June 7, 1990, at least four days are available following a loss of SFC before boiling can occur in the SFP, and almost eight additional days are available before 6.75 Feet of SFP water could boil off. A chlorine gas spill would not prevent operator actions to restore a loss of SFC or SFP inventory before enough SFP heatup and boiling could occur that would uncover spent fuel assemblies or threaten spent fuel integrity.
	Because the plant is shutdown, the consequences of a credible accident in the PDM are minimal, and several days are available for operators to take actions following a loss of SFC, the chlorine issue is no longer a Nuclear Safety issue but an industrial safety issue. Also, NUREG-0737, Action Item III.D.3.4, Control-Room Habitability Requirements, states that these requirements apply to all operating reactors. Therefore, a chlorine specification is not required to protect the health and safety of the public in the PDM, and Specifications associated with chlorine gas detectors or limits are not required to be included in the PDTS.
	Specifications 3.13.1 and 3.13.4 address the Operability of the Reactor Building Purge Exhaust Filter System in relation to continued reactor operation and containment integrity. In the PDM, reactor operation is not allowed and containment integrity is not required; therefore, these Specification requirements are not included in the PDTS.
	The requirements for operating the Reactor Building Purge Exhaust Filter System as a Gaseous Radwaste Treatment System is addressed in the evaluation for Specification 3.18.4, Gaseous Radwaste Treatment. Specification 3.18.4 is being removed from the Technical Specifications per Generic Letter 89-01 and its requirements relocated to the Offsite Dose Calculation Manual (ODCM).

Technical Specification Number	Discussion
3.13 (Cont.)	Specification 3.13.2 provides the requirements for the Operability of the Auxiliary and Spent Fuel Building Filter System when the spent fuel in the SFP has decayed less than 30 days. Since the spent fuel has decayed for greater than 30 days, this Specification is not applicable and is not included in the PDTS. The requirements for operating the Auxiliary and Spent Fuel Building Filter System in the PDM is associated with the design objectives of 10 CFR 50, Appendix I. These requirements are contained in Specification 3.18.4 and are addressed in the evaluation for Specification 3.18.4. The Ventilation Exhaust Treatment System portion of Specification 3.18.4 is removed from the Technical Specifications per Generic Letter 89-01 and relocated to the ODCM. (See the evaluation for Specification 3.18.4 on page 21 of this SAR.)
3.14	Specifications 3.14 were Fire Protection Technical Specifications that were previously deleted pursuant to Generic Letter 88-12 and License Amendment No. 115.

The following list of Specifications have been removed from the Technical Specifications pursuant to the requirements of Generic Letter 89-01 and placed in the ODCM as they currently exist, except as noted in the following evaluations. A revised ODCM that incorporates the existing Radiological Effluent Technical Specification (RETS) requirements applicable in the PDM is included as Attachment IV to the PA-182, Revision 2 submittal package for NRC use as a reference.

3.15	Radioactive Liquid Effluent Monitoring Instrumentation
3.16	Radioactive Gaseous Effluent Monitoring Instrumentation
3.17.1	Liquid Effluents Concentration
3.17.2	Liquid Effluents Dose
3.17.4	Liquid Effluent Radwaste Treatment
3.18.2	Gaseous Effluents Dose-Noble Gases
3.18.3	Gaseous Effluents Dose-Iodine-131, Iodine-133, Tritium and Radioactive Materials in Particulate Form
3.18.4	Gaseous Radwaste Treatment (Ventilation Exhaust Treatment Systems only)
3.25	Fuel Cycle Dose

Technical Specification Number	Discussion
3.17.3 (D3/4.6)	This Specification provides radioactive material content limits for certain outdoor liquid holdup tanks, is retained in the Technical Specifications as required by Generic Letter 89-01, and is included in the PDTS as Specification D3/4.6.

Technical Specification Number	Discussion
3.18.1	The Site Boundary for Gaseous Effluents (PDTS Figure D5.1-2) will be used in lieu of the Exclusion Area currently used for evaluating the dose rate limits of Specification 3.18.1. This change does not affect the current dose calculation methods specified in the ODCM for gaseous effluents, is consistent with the other gaseous effluent Technical Specifications (3.18.2, 3, and 4), and is consistent with NUREG-0133, Preparation Of Radiological Effluent Technical Specifications For Nuclear Power Plants. Specification 3.18.1 is relocated to the ODCM per Generic Letter 89-01.
3.18.4	The Ventilation Exhaust Treatment System (VETS) operating requirements stated in Specification 3.18.4 are relocated to the ODCM in accordance with Generic Letter 89-01. However, Specification 3.18.4, Gaseous Radwaste Treatment, addresses both the Waste Gas System (WGS) and the VETS. In the PDM, the WGS is not required to perform its intended function or meet the design objectives of 10 CFR 50, Appendix I; therefore, operating requirements for the WGS are not included in the PDTS. The WGS is designed to collect and hold for eventual release fission product gases generated in the Reactor Coolant System (RCS) during reactor operation. This design meets the design objectives of 10 CFR 50, Appendix I for an operating plant. Now that the reactor has been defueled and the RCS depressurized and vented to atmosphere for an extended period of time, essentially all the fission product gases contained in the RCS and collected in the WGS have been released through the WGS. With negligible amounts of radioactive gases available for discharge through the WGS in the PDM, the WGS is not required to be used as a Gaseous Radwaste Treatment (GRT) system to help ensure to 10 CFR 50, Appendix I guidelines are met in the PDM. This Appendix I design objective is met through the use of the VETS as GRT systems, because the composition of gaseous effluent in the PDM will be almost exclusively particulate matter with no Iodine. Therefore, in the PDM the WGS is not required to collect and hold gases for Appendix I considerations, and the GRT requirements for the WGS are not included in the PDTS or the ODCM. For further discussion on the WGS in regards to explosive gas mixtures, see the following evaluation for Specifications 3.18.5 and 3.24.

Technical Specification Number	Discussion
3.18.5 3.24	Gas Storage Tanks Explosive Gas Mixture

Specifications 3.18.5 and 3.24 provide radioactivity content and explosive gas mixture limits for the Waste Gas Decay Tanks (WGDTs), which are components of the WGS. The WGDTs are used to store fission product gases and other off-gases from the Reactor Coolant System (RCS) produced during reactor operation and removed from the RCS by the Makeup and Purification System. During normal plant operation, hydrogen is used as the cover gas in the Makeup Tank to provide a means of oxygen scavenging in the RCS. Now that the reactor has been defueled and the RCS depressurized and vented to atmosphere for an extended period of time, the fission product gases and hydrogen normally entrained in the RCS have been released through the WGDTs.

Any remaining radioactive or other gases normally collected in the WGDTs will be discharged through the Auxiliary Building Stack Ventilation System without being held up in the WGDTs. Thus, no potential exists for accumulating an explosive gas mixture in the WGDTs. Also, the quantity of radioactive gases available to the WGS in the PDM is minimal. The design basis for the activity content limit on the WGDTs assumes there is fuel in the reactor and an excessively high RCS specific activity. This design basis condition is not applicable in the PDM. Therefore, it is not necessary to hold up radionuclides in the WGDTs or monitor the concentration of oxygen in the WGDTs during the PDM. The requirements associated with these Specifications are not applicable in the PDM and are not included in the PDTS.

3.21 This Specification provides requirements for Solid Radioactive Wastes and has been removed from the Technical Specifications, pursuant to the requirements of Generic Letter 89-01, and placed in the Process Control Program (PCP). The PCP was included as part of the original PA-182 submittal package for NRC use as a reference. A copy of the PCP is included as Attachment VI for completeness of this PDTS submittal.

3.22 Radiological Environmental Monitoring
3.23 Land Use Census

Specifications 3.22 and 3.23 have been removed from the Technical Specifications pursuant to the requirements of Generic Letter 89-01 and placed in the Radiological Environmental Monitoring Program (REMP) Manual. The REMP Manual that was included as part of the original PA-182 submittal package for NRC is editorially modified and resubmitted as Attachment VIII to this submittal package (PA-182, Rev. 2) for NRC use as a reference. Attachment VIII replaces the original submittal of the REMP Manual in its entirety.

Technical Specification Number	Discussion
3.26	This Specification provides Interlaboratory Comparison Program requirements that are applicable to the REMP. This requirement has been removed from the Technical Specifications pursuant to Generic Letter 89-01 and placed in the REMP Manual.
3.27	Specification 3.27 provides requirements for the Nuclear Service Electrical Building (NSEB) Emergency Heating Ventilation and Air Conditioning (HVAC) System. This system provides a backup emergency HVAC system for the NSEB and the electrical equipment contained within the building. The requirements of this Specification are applicable in the Heatup through Power Operations modes and are not applicable in the PDM. With the plant shutdown and the reactor defueled, the heat load in the NSEB is significantly reduced since most of the equipment supplied from buses in this building is deenergized. A significant period of time is available to allow restoration of the normally operating HVAC system should the system become unavailable. A Specification for this system is not required in the PDM and is not included in the PDTS.
3.28	This Specification provides requirements for the TDI Diesel Generator Control Room Essential Ventilation System and is applicable in the Heatup through Power Operations modes. The emergency power sources are not required to be maintained Operable in the PDM. The definition for Operable is modified to exclude emergency power source requirements. Based on the LOOP analysis and the requirements associated with a plant in the PDM, Specification 3.28 is not applicable in the PDM and is not included in the PDTS.
3.29	Specification 3.29, Meteorological Monitoring Instrumentation, is not included in the PDTS because the defueled condition Emergency Plan and Effluent Control Program do not require information from this instrumentation to provide protection of the public health and safety. The original bases for this specification was to ensure the availability of sufficient meteorological data for estimating potential radiation doses to the public as a result of routine or accidental releases of radioactive material to the atmosphere. This capability was required to evaluate the need for initiating protective measures to protect the health and safety of the public.

Technical Specification Number	Discussion
3.29 (Cont.)	As a result of a public vote, Rancho Seco was shut down on June 7, 1989, and completely defueled on December 8, 1989. Also, Confirmatory Order dated May 2, 1990, prevents the District from moving fuel into the Reactor Building without prior NRC approval. As discussed in the granting of the partial exemption to 10 CFR 55 (NRC to SMUD letter dated May 16, 1990, TAC No. 75520) and on page 2 of this SAR, the design basis accidents considered credible in the PDM are reduced to a Loss of Offsite Power and a Fuel Handling Accident.

During a routine or accidental release of gaseous radioactive material, the magnitude of the offsite doses is dictated primarily by the source term and the atmospheric dispersion coefficient, X/Q. The total curies of noble gases and iodines released is directly proportional to the offsite doses. With the exception of Krypton-85, these isotopes have half lives of only a few days and therefore have essentially all decayed away now that the reactor has been shut down for well over a year. Krypton-85 is now the predominant isotope in the gaseous source term. The accident analyses performed for the PDM concluded that the Krypton-85 source term is so small during a design basis accident that the total whole-body dose to an individual at the site perimeter would be only a few millirems during the entire passage of the plume. The resulting total thyroid dose would be only a small fraction of a millirem due to decay of the short lived iodines. Thus, the consequences of accidents considered credible in the defueled condition will not exceed the plume exposure Protective Action Guidelines. These small doses result even though the accident analyses assume default Class F meteorology (X/Q = 6.33E-4 sec/m^3).

Because of the extremely small source term that exists in the defueled condition, it is expedient and conservative to use a default X/Q value in calculations involving accidental releases of radioactive gaseous effluent, instead of relying on meteorological monitoring instrumentation to provide the data needed to calculate actual X/Q values. Also, the District has established a direct correlation between Spent Fuel Pool area radiation monitor readings and the dose consequences at the site boundary following an accident in the spent fuel storage building. This correlation is used in the defueled condition Emergency Plan to evaluate the consequences of an accidental airborne release from the spent fuel storage building, employing the standard accident analysis default X/Q value of 6.33E-4 sec/m^3. Therefore, the extended decay time of the irradiated fuel in the Spent Fuel Pool has obviated the need for meteorological monitoring instrumentation for emergency preparedness purposes in the PDM.

Technical Specification Number	Discussion
3.29 (Cont.)	The meteorological monitoring instrumentation provides the data needed to determine the actual X/Q and D/Q values used to calculate the dose impact of normal gaseous effluent. This calculated dose impact is reported to the NRC on a semiannual basis. Because the plant has been shut down since June 1989, the gaseous effluent source term is extremely small, thus, it is possible to use conservative default X/Q and/or D/Q values to calculate routine or accidental releases of radioactivity. For example, as discussed above, it is no longer necessary to retrieve data from the meteorological monitoring instrumentation during an accident scenario. Instead, it is conservatively assumed that default Class F meteorology exists with corresponding X/Q = 6.33E-4 sec/m^3. This conservative X/Q value was used in the evaluation that determined the design basis defueled condition accident at Rancho Seco results in a maximum whole body dose of only a few millirems.

Likewise, for routine or normal releases of radioactive gaseous effluent, data from meteorological monitoring instrumentation is not needed in the PDM. Routine radioactive gaseous effluent has reduced to almost zero since the plant was shut down in June 1989. No measurable levels of iodines or noble gases are present in gaseous effluent. Only very small amounts of Tritium are detected in gaseous effluent discharged from the plant. Instead of using actual X/Q and D/Q values to calculate the dose impact of normal gaseous effluent to meet semiannual reporting requirements, use of a conservative default X/Q value (1.00E-4 sec/m^3) and D/Q value (1.00E-6 m^{-2}), based on historical data, is proposed. Since X/Q and D/Q is directly proportional to the calculated dose, using a conservative X/Q and D/Q value will ensure conservative reporting of the calculated dose impact of normal gaseous effluent resulting from operation of the facility in the PDM. Because the source term for normal gaseous effluent in the PDM is extremely small, use of a conservative X/Q and D/Q will result in conservative reporting of the normal gaseous effluent dose, but will not affect the plant's ability to meet the dose guidelines of 10 CFR 50, Appendix I.

To illustrate this last point, the following evaluation of gaseous effluent dose data, X/Q data, and D/Q data is provided.

The post shutdown gaseous effluent quarterly dose data reported in the July-December 1989 and January-June 1990 Semiannual Radioactive Effluent Release Reports is presented below.

Technical Specification Number	Discussion
3.29 (Cont.)	

	3rd Qtr* 1989	4th Qtr* 1989	1st Qtr* 1990	2nd Qtr* 1990
Tritium, Iodine, Particulate:				
Maximum Organ Dose	2.91E-2	4.31E-2	2.18E-2	1.71E-2
Percent Tech Spec Limit	3.88E-1	5.75E-1	2.91E-1	2.28E-2
Noble Gas:				
Gamma Air Dose	1.74E-4	1.51E-5	0	0
Percent Tech Spec Limit	3.48E-3	3.02E-4	0	0
Beta Air Dose	5.17E-4	1.72E-3	0	0
Percent Tech Spec Limit	5.17E-3	1.72E-2	0	0

* The Maximum Organ Dose is in mrem and the Gamma and Beta Air Dose is in mrad.

These calculated results are based on quarterly averaged X/Q and D/Q values generated from meteorological monitoring instrumentation data collected during the applicable period. The third quarter 1989 through second quarter 1990 X/Q and D/Q values used are 2.30E-5 sec/m^3 and 4.60E-7 m^{-2}, 2.30E-5 sec/m^3 and 3.40E-7 m^{-2}, 2.20E-5 sec/m^3 and 4.10E-7 m^{-2}, and 1.90E-5 sec/m^3 and 3.00E-7 m^{-2}, respectively.

If the proposed conservative default X/Q and D/Q values (1.00E-4 sec/m^3 and 1.00E-6 m^{-2}, respectively) are used instead of the X/Q and D/Q values calculated from actual meteorological data, the following post shutdown gaseous effluent quarterly doses result.

Technical Specification Number	Discussion
3.29 (Cont.)	

	3rd Qtr* 1989	4th Qtr* 1989	1st Qtr* 1990	2nd Qtr* 1990
Tritium, Iodine, Particulate:				
Maximum Organ Dose	2.25E-1	2.59E-1	2.06E-1	1.59E-1
Percent Tech Spec Limit	3.00E+0	3.45E+0	2.75E+0	2.12E+0
Noble Gas:				
Gamma Air Dose	7.57E-4	6.57E-5	0	0
Percent Tech Spec Limit	1.51E-2	1.31E-3	0	0
Beta Air Dose	2.25E-3	7.48E-3	0	0
Percent Tech Spec Limit	2.25E-2	7.48E-2	0	0

* The Maximum Organ Dose is in mrem and the Gamma and Beta Air Dose is in mrad.

By comparing the quarterly dose tables presented above, it is seen that use of a conservative default X/Q and D/Q value in the shutdown condition would result in conservative reporting of dose, but would not affect the plant's ability to maintain releases within the 10 CFR 50, Appendix I dose guidelines.

The previous 10 quarters of X/Q and D/Q values calculated from actual meteorological data range from a high of 2.85E-5 sec/m^3 to a low of 1.80E-5 sec/m^3 and a high of 4.60E-7 m^{-2} to a low of 2.61E-7 m^{-2}, respectively. The ratio of the proposed conservative default X/Q and D/Q values to actual quarterly averaged X/Q and D/Q values range from 3.51 to 5.56 and 2.17 to 3.83, respectively. Since X/Q and D/Q is directly proportional to the calculated dose, this range of ratios represents the degree of conservatism associated with the proposed default X/Q.

Because of (1) the extremely small releases associated with the plant in the shutdown condition, (2) the conservative impact the proposed X/Q and D/Q values have on reported gaseous effluent doses, and (3) the lack of impact on the plant's ability to maintain the reported normal gaseous effluent doses within the 10 CFR 50, Appendix I dose guidelines, it is expedient and conservative to use the proposed conservative default X/Q (1.00E-4 sec/m^3) and D/Q (1.00E-6 m^{-2}) values. Therefore, based on the above analysis, meteorological monitoring instrumentation is

Technical Specification Number	Discussion
3.29 (Cont.)	not required to support the derivation of the X/Q and D/Q values used in the calculations for normal gaseous effluent dose impact, and a Specification addressing the Operability requirements of meteorological monitoring instrumentation is not required to support the Effluent Control Program in the PDM. In summation, the extremely small source term that currently exists at Rancho Seco justifies the use of default X/Q and D/Q values in the evaluation of both routine and accidental releases of radioactivity. Using a conservative X/Q and D/Q value in normal gaseous effluent dose calculations and a standard default X/Q value in accident analysis dose evaluations, preempts the need to maintain meteorological monitoring instrumentation in the PDM; therefore, a meteorological monitoring instrumentation Specification is not required to be included in the PDTS. If meteorological information is desired in relation to an event at Rancho Seco during the PDM, it may be obtained from the National Weather Service in Sacramento.
3.30	This Specification provides requirements for the Hydrogen Recombiners which control hydrogen gas inventory in the containment following a LOCA. The Hydrogen Recombiners are required to be Operable when the reactor is subcritical by less than 1 percent $\Delta k/k$. Since the reactor is defueled, the Hydrogen Recombiners are not required in the PDM and this Specification is not included in the PDTS.
4.0.1 (D4.0.1)	This Specification is included in the PDTS as Specification D4.0.1. An editorial change is made to reflect the single mode of applicability of the PDTS; the PDM.
4.0.2 (D4.0.3)	This Specification is editorially modified to reflect the implementation of Generic Letter 89-14 and NRC comments, and is included in the PDTS as Specification D4.0.3.
4.0.3 (D4.0.4)	The requirements of this Specification are editorially modified to reflect the single mode of applicability of the PDTS. These requirements are included in the PDTS as Specification D4.0.4.
4.0.4	This Specification provides an extension for certain Appendix A surveillances until the Cycle 8 refueling outage. These surveillances are not applicable in the PDM; therefore, this Specification is not included in the PDTS.

Technical Specification Number	Discussion
4.1 Tables 4.1-1, 4.1-2, 4.1-3	This Specification provides surveillance and testing requirements for instrumentation associated with RPS, SFAS, Process Instrumentation, and Emergency Shutdown Instrumentation. Most of this instrumentation is required for reactor operation and transient monitoring, and accident identification and mitigation for an operating nuclear power plant, and is not applicable in the PDM. The surveillance requirements listed in Tables 4.1-1, 4.1-2, and 4.1-3 are evaluated below for applicability in the PDM and inclusion in the PDTS.

The following is an evaluation of the Table 4.1-1 items which were previously determined to be applicable in the defueled condition.

Table 4.1-1, Items 44a, 44b, and 45, Process, Area, and Emergency radiation monitoring instrumentation surveillances, are dispositioned as follows:

a) Surveillance and testing requirements for those radiation monitors required by the Standard RETS (NUREG-0472) and associated with radioactive effluent control and monitoring have been relocated to the ODCM in accordance with the requirements of Generic Letter 89-01. (See the evaluation for Specifications 3.15, 3.16, 3.17 and 3.18 on pages 20 and 21 of this SAR.)

b) The SFP area radiation monitor surveillance requirements are included in the PDTS as part of Specification D3/4.4.

c) The remaining radiation monitors are used for general area and process monitoring, are not included in the STS, are controlled by administrative and technical procedures, and are not included in the PDTS.

d) The surveillance requirements for Item 45 of Table 4.1-1, Emergency Plant Radiation Instruments, are not included in the PDTS because these requirements are included in an implementing procedure for the approved Emergency Plan. Changes to this procedure would require both a 10 CFR 50.59 and 10 CFR 50.54(q) review to determine if an Unreviewed Safety Question or a reduction in the effectiveness of the Emergency Plan would result. Also, the Nuclear Safety Review and Audit Committee and the AGM, Nuclear is required by the PDTS (Specifications D6.5.1.7d and D6.5.3a, respectively) to review all Emergency Plan implementing procedure changes. Therefore, the surveillance requirements specified in Table 4.1-1, Item 45 are not required in the PDM and are not included in the PDTS.

Technical Specification Number	Discussion
4.1 Tables 4.1-1, 4.1-2, 4.1-3 (Cont.)	Table 4.1-1, Item 46, Environmental Air Monitors, has been relocated to the REMP Manual in accordance with Generic Letter 89-01.

Table 4.1-1, Item 47, Strong Motion Accelerometer, is used to provide information for plant shutdown and cooldown following a Safe Shutdown Earthquake or Design Basis Earthquake. Since the plant is already shut down and seismic data is available from State and Federal agencies, the accelerometer is not required in the PDM. Therefore, the surveillance requirement for this instrumentation is not included in the PDTS.

Table 4.1-1, Item 51, SFP Level, is included in the PDTS as part of Specification D3/4.1.

The following is an evaluation of the Table 4.1-1 items which were previously determined to not be applicable in the defueled condition.

The RPS and SFAS related surveillance requirements listed in Table 4.1-1, Items 1 through 27, apply to an operating nuclear power plant, are not applicable in the PDM, and are not included in the PDTS.

The reactor operation, transient, and accident monitoring instrumentation and Emergency Core Cooling System process instrumentation surveillance requirements addressed in Table 4.1-1, Items 28 through 43, 49, 50, 52 through 56, 58 through 76, and 84 through 90 are not applicable in the PDM and are not included in the PDTS. No surveillances are associated with Items 48 and 77 through 81.

Table 4.1-1, Item 57, Voltage Protection, is not required because if degraded voltage or a LOOP is experienced, several hours are available to regain electric power and re-establish SFC (see the evaluation on page 3 of this SAR). The emergency power sources are not required in the PDM; therefore, Item 57 is not included in the PDTS.

Table 4.1-1, Items 82 and 83, Spray Pond Water Temperature and Level, is not included in the PDTS based on the LOOP analysis and the amount of time available to restore SFC.

The following is an evaluation of the Table 4.1-2 items which were previously determined to not be applicable in the defueled condition.

Technical Specification Number	Discussion
4.1 Tables 4.1-1, 4.1-2, 4.1-3 (Cont.)	Table 4.1-2, Items 1 through 4, 6, 7, 10, and 12 through 19 are surveillances that are required for a plant with fuel in the reactor or a plant that intends to operate. In the PDM, no fuel is allowed in the reactor building and no intent to operate the plant exists. Therefore, these surveillances are not included in the PDTS.

The following is an evaluation of the Table 4.1-2 surveillance requirements which were previously determined to be applicable in the defueled condition.

Table 4.1-2, Item 5, Refueling System Interlocks, is included in the PDTS as part of Specification D3/4.3 to require testing of the Spent Fuel Building fuel handling system interlocks only. The Applicability Statement for this surveillance ensures completion of a Functional Test within 7 days prior to fuel handling and once per 7 days thereafter during fuel handling operations. A definition for Functional Test is provided in the PDTS as Specification D1.4.5.

Table 4.1-2, Item 8, Charcoal and High Efficiency Filters, is not included in the PDTS. Because the plant has been shut down since June 1989, iodine levels in the plant have decayed away to essentially zero and no production mechanism for radioactive iodine exists in the PDM. Therefore, charcoal filtration units are not required in the PDM. Also, the radioactive gaseous effluent requirements which address the operation of the HEPA Filters (Specifications 3.18.4 and 4.22.4) have been relocated to the ODCM in accordance with Generic Letter 89-01. For further discussions on the requirements of the charcoal and HEPA Filters in the PDM, see the evaluations for Specifications 3.13, 3.18.4, and 3.18.5 on pages 20 through 22 of this SAR. Also, SMUD to NRC letter AGM/NUC 90-095 dated March 21, 1990, addresses the operation of the HEPA/Charcoal Filtration units in the defueled condition.

The Table 4.1-2, Item 9, Fire Pumps and Power Supplies, was relocated to the Fire Protection Plan pursuant to Generic Letter 88-12 and Rancho Seco License Amendment No. 115.

Table 4.1-2, Item 11, SFC System, requires functional testing of the SFC system each refueling interval prior to fuel handling. This requirement is met by continuously maintaining the SFP temperature below 140 degrees Fahrenheit. The intent of the Specification is to test the SFC system prior to the addition of more spent fuel to the SFP. Because no more Spent Fuel will be added to the SFP in the PDM, the heat load in the SFP will continue to decrease from a calculated value of 3.60 million BTU/hour as of January 3, 1990 to 1.86 million BTU/hour on June 7, 1991. This compares to a design capacity of the SFC system of

Technical Specification Number	Discussion
4.1 Tables 4.1-1, 4.1-2, 4.1-3 (Cont.)	8.76 million BTU/hour. Extraordinary system degradation would have to occur for the system to not meet its intended function. Therefore, this Functional Test surveillance requirement is met by verifying the SFP temperature is maintained below 140°F. Specification D3/4.2, Surveillance Requirement D4.2 is sufficient to ensure the SFC system is functioning; therefore, this requirement is already sufficiently addressed and is not included in the PDTS. The Table 4.1-3 surveillance requirements were previously evaluated as not applicable in the defueled condition. The following is the justification for not including the Table 4.1-3 surveillance requirements in the PDTS. Table 4.1-3, Items 2,4, and 6, Maintenance of a Minimum Boron Concentration in the Borated Water Storage Tank, SFP, and Concentrated Boric Acid Tank, respectively, are not required in the PDM since borated water is not needed to maintain an adequate shutdown margin in the SFP (see evaluation on page 4 of this SAR). Therefore, these requirements are not included in the PDTS. Table 4.1-3, Items 1, 3, 5, 7 and 8 are applicable only for an operating plant with fuel in the reactor. In the PDM, these surveillance requirements are not applicable; therefore, they are not included in the PDTS.
4.2	Specifications 4.2.2.1 through 4.2.2.5 provide in-service inspection and testing requirements for certain safety-related components classified as ASME Code Class 1, 2, and 3 in accordance with 10 CFR 50.55a and ASME Section XI. The inspections and tests identified verify the integrity over time of critical safety-related systems and other systems whose integrity is required for control of the release of fission products during and following a severe reactor accident with fuel damage. The testing requirements for pumps and valves are established to verify equipment is capable of meeting its design requirements. The systems remaining in service at Rancho Seco are experiencing a continual decrease in required capability as a function of time due to the radioactive decay of the spent fuel. The systems required to be Operable in the PDM operate at relatively low pressure and temperature when compared to their design limits. Therefore, failures and degradation which can be experienced at high temperature and pressure is not expected in systems required to be Operable in the PDM. The systems addressed by ASME Section XI for Rancho Seco that are required to be included in a Section XI inspection and testing program are not required to be Operable in the PDM.

Technical Specification Number	Discussion
4.2 (Cont.)	ASME Section XI, code section IWA-2400(c) provides for the deferral of inspections during extended plant outages. ASME Section XI, code section IWP-3400 allows the suspension of pump testing during plant shutdowns. ASME Section XI, code section IWV-3410 allows the suspension of valve testing when a system is inoperable or is not required to be Operable. Rancho Seco was shut down in June 1989, following a public vote, and defueled in December 1989. Being in the PDM constitutes a permanent plant shutdown. Also, no systems which are a part of the ASME Section XI testing program are required to be Operable in the PDM. Therefore, no specifications are required to be included in the PDTS which address ASME Section XI inspection and testing requirements.
	The District has no intent to operate Rancho Seco as a nuclear facility. NRC Order dated May 2, 1990, does not allow movement of fuel into the reactor building, and therefore, obviates equipment otherwise required to be inspected or tested per ASME Section XI. Components or systems required to be tested or inspected under ASME Section XI or that are part of the NRC approved Rancho Seco inspection and testing program are not required to be Operable in the PDM. Therefore, no inspections or tests are required to meet ASME Section XI or 10 CFR 50.55a requirements. ASME Section XI and 10 CFR 50.55a are met without performance of inspections or testing. A Specification which addresses these requirements is not required in the PDM and should not be included in the PDTS. Table 2 on pages 57 through 70 of this SAR provides the list of components included in the latest, approved Rancho Seco ASME Section XI testing program and a determination of which components are required to be tested according to the current Technical Specifications and the PDTS.
	Specification 4.2.3 is the surveillance requirement for Specification 3.1.6, which is only applicable above Cold Shutdown. Monitoring the RCS for leakage is unnecessary in the PDM because the reactor is defueled. This surveillance is not included in the PDTS. See the evaluation for Specification 3.1 on page 10 of this SAR.
4.3	The verification of RCS integrity prior to returning to criticality is not applicable in the PDM and is not included in the PDTS.

Technical Specification Number	Discussion
4.4	Specifications 4.4.1.1 through 4.4.2.6 provide testing, inspection, and inservice surveillance requirements which monitor the structural and leakage integrity of the Reactor Building. In the PDM, there are no credible accidents that require Reactor Building integrity. Maintaining Reactor Building integrity as specified in these surveillance requirements is no longer required to provide reasonable assurance that activities conducted at the facility will not result in undue risk to the health and safety of the public. Therefore, these surveillance requirements are not required in the PDM and are not included in the PDTS.
4.5	Specifications 4.5.1 through 4.5.3 provide testing requirements to verify proper operation of the Emergency Core Cooling and Reactor Building Spray systems. With the reactor defueled, the need for these systems no longer exists. Therefore, these surveillance requirements are not included in the PDTS. The DHRS leakage testing requirement in Specification 4.5.3 is predicated on assuring that the DHRS will operate to cool the reactor for approximately 200 days and will minimize the dose due to leakage from this system following a loss of coolant accident (LOCA). In the PDM, a LOCA as defined in the USAR is not credible; therefore, this requirement is not included in the PDTS.
4.6	The emergency power systems are not required when the plant is shut down, thus the surveillances for these systems are not required in the PDM and are not included in the PDTS. See the evaluation for the LOOP event and Specification 3.7 on pages 3 and 12 of this SAR, respectively.
4.7	Specifications 4.7.1.1 through 4.7.2.3 provide testing requirements for the control rods. Since the reactor is defueled, these Specifications are not applicable in the PDM and are not included in the PDTS.
4.8	Specifications 4.8.1 through 4.8.5 provide testing requirements for the Auxiliary Feedwater System. Since the reactor is defueled, there is no need to provide a feedwater supply for the steam generators. Therefore, these Specifications are not included in the PDTS.
4.9	Specification 4.9 requires evaluation of reactivity anomalies that occur during plant operation. Since the reactor is defueled in the PDM, this Specification is not applicable and is not included in the PDTS.
4.10	This surveillance requirement addresses the CR/TSC Emergency Filtering System, is not applicable in the PDM, and is not included in the PDTS. See the evaluation for Specification 3.13 on pages 18 through 20 of this SAR.

Technical Specification Number	Discussion
4.11	Specification 4.11 is the surveillance requirements for the Reactor Building Purge Exhaust Filtering System. See the evaluation for Specifications 3.13.1, 3.13.4, 3.18.4, and Table 4.1-2, Item 8 on pages 18 through 21 and 31 of this SAR.
4.12	Specification 4.12 is the surveillance requirements for the Auxiliary and Spent Fuel Building Filter Systems. See the evaluation for Specifications 3.13.2, 3.18.4, and Table 4.1-2, Item 8 on pages 20, 21, and 31 of this SAR.
4.13	This Specification provides inspection requirements for specific high energy lines outside the Reactor Building. Since the high energy systems (Main Feedwater and Main Steam) associated with this inspection requirement are not in service in the PDM, this Specification is not applicable and is not included in the PDTS.
4.14	Specification 4.14 is the surveillance requirement for Specification 3.12, Snubbers. These requirements are not included in the PDTS. See the evaluation for Specification 3.12 on page 18 of this SAR.
4.15	Specification 4.15, Radioactive Materials Sources, is rewritten in accordance with the B&W Standard Technical Specifications (NUREG-0103) and is included in the PDTS as Specification D3/4.7. Proposed Specification D3/4.7 reflects the sealed source leak testing requirements applicable for a plant in the PDM, and ensures leakage from byproduct, source, and special nuclear radioactive material sources will not exceed allowable intake values. These requirements meet the general sealed source leak testing requirements of 10 CFR 35.39. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on the 10 CFR 70.39(c) limits for Plutonium.
4.16	This Specification was previously deleted.
4.17	This Specification provides testing requirements for the Steam Generators. Since the Steam Generators are not in service in the PDM, these requirements are not included in the PDTS.
4.18	These Fire Protection surveillance requirements were previously deleted. See the evaluation for Specification 3.14 on page 20 of this SAR.
4.19	This is the surveillance requirement for Specification 3.15. See the evaluation for Specification 3.15 on page 20 of this SAR.
4.20	This is the surveillance requirement for Specification 3.16. See the evaluation for Specification 3.16 on page 20 of this SAR.

Technical Specification Number	Discussion
4.21.1	This is the surveillance requirement for Specification 3.17.1. See the evaluation for Specification 3.17.1 on page 20 of this SAR.
4.21.2	This is the surveillance requirement for Specification 3.17.2. See the evaluation for Specification 3.17.2 on page 20 of this SAR.
4.21.3	This is the surveillance requirement for Specification 3.17.3. See the evaluation for Specification 3.17.3 on page 20 of this SAR.
4.21.4	This is the surveillance requirement for Specification 3.17.4. See the evaluation for Specification 3.17.4 on page 20 of this SAR.
4.22.1	This is the surveillance req[illegible]ement for Specification 3.18.1. See the evaluation for Specif[illegible]1 on page 21 of this SAR.
4.22.2	This is the surveillance requirement for Specification 3.18.2. See the evaluation for Specification 3.18.2 on page 20 of this SAR.
4.22.3	This is the surveillance requirement for Specification 3.18.3. See the evaluation for Specification 3.18.3 on page 20 of this SAR.
4.22.4	This is the surveillance requirement for Specification 3.18.4. See the evaluation for Specification 3.18.4 on page 21 of this SAR.
4.22.5	This is the surveillance requirement for Specification 3.18.5. See the evaluation for Specification 3.18.5 on page 22 of this SAR.
4.23	This Specification was previously deleted.
4.24	This Specification was previously deleted.
4.25	This is the surveillance requirement for Specification 3.21. See the evaluation for Specification 3.21 on page 22 of this SAR.
4.26	This is the surveillance requirement for Specification 3.22. See the evaluation for Specification 3.22 on page 22 of this SAR.
4.27	This is the surveillance requirement for Specification 3.23. See the evaluation for Specification 3.23 on page 22 of this SAR.
4.28	This is the surveillance requirement for Specification 3.24. See the evaluation for Specification 3.24 on page 22 of this SAR.
4.29	This is the surveillance requirement for Specification 3.25. See the evaluation for Specification 3.25 on page 20 of this SAR.
4.30	This is the surveillance requirement for Specification 3.26. See the evaluation for Specification 3.26 on page 23 of this SAR.

Technical Specification Number	Discussion
4.31	This is the surveillance requirement for Specification 3.27. See the evaluation for Specification 3.27 on page 23 of this SAR.
4.32	This is the surveillance requirement for Specification 3.28. See the evaluation for Specification 3.28 on page 23 of this SAR.
4.33	This Specification was previously deleted.
4.34	This is the surveillance requirement for Specification 3.29. See the evaluation for Specification 3.29 on pages 23 through 28 of this SAR.
4.35	This is the surveillance requirement for Specification 3.30. See the evaluation for Specification 3.30 on page 28 of this SAR.
5.1 (D5.1)	Figures 5.1-1 and 5.1-2, Exclusion Area (EA) and Low Population Zone (LPZ), respectively, are combined into a single Figure (PDTS Figure D5.1-1) titled Emergency Planning Zone (EPZ). This new boundary is established in accordance with the defueled condition Emergency Plan. The EPZ is reduced to the Industrial Area outlined in PDTS Figure D5.1-1. The size of the EPZ boundary is based on the minimal consequences that result from the few accidents considered credible in the PDM. At this boundary, calculated radioactive exposures do not exceed the plume exposure Protective Action Guidelines of EPA 520/1-75-001-A, January 1990, Manual of Protective Actions for Nuclear Incidents, for the worst case defueled condition design basis accident (the dropped fuel assembly accident). The EA and LPZ are acceptability criteria used by the NRC to determine suitability of proposed reactor sites in relation to 10 CFR 100 release limits. The EA and LPZ are not applicable in the PDM, because the dose consequence of the dropped fuel assembly accident in the PDM is an extremely small fraction of the 10 CFR 100 limits (see the evaluation on page 2 of this SAR), and thus are replaced by the EPZ in the PDTS. The titles of Figures 5.1-3 and 5.1-4 are modified in accordance with the Standard Radiological Effluent Technical Specifications (NUREG-0472) to clarify the application and significance of the two effluent boundaries. Also, the two figures are renumbered D5.1-2 and D5.1-3 and included in the PDTS. The gaseous effluent boundary (PDTS Figure D5.1-2) applies to gaseous effluents released from the site under normal conditions and is used to evaluate both 10 CFR 20 and 10 CFR 50, Appendix I compliance. The liquid effluent boundary (PDTS Figure D5.1-3) identifies the boundary at which radiological liquid effluent actually leaves the Industrial Area. The dose accountability points for meeting the 10 CFR 50, Appendix I guidelines remains at the A and B Regenerant

Technical Specification Number	Discussion
5.1 (D5.1) (Con.)	Hold-Up Tanks, and the concentration accountability points for determining 10 CFR 20, Appendix B compliance remains at the Retention Basins. Under accident conditions the EPZ boundary is used to evaluate the potential consequences of an accident and the corresponding response and notification actions required in accordance with the defueled condition Emergency Plan. The EPZ boundary is the same as the liquid effluent boundary, and more conservative than the gaseous effluent boundary.
5.2	The two systems addressed in this Specification, Reactor Building and Reactor Building Isolation, are not required to be Operable in the PDM, thus this Specification is not included in the PDTS.
5.3	This section describes the Reactor. Since the Reactor is defueled, this Specification is no longer applicable and is not included in the PDTS.
5.4 (D5.2)	The descriptions of spent fuel storage in the Spent Fuel Storage Building have been included in the PDTS as Specification D5.2. The remainder of this Specification addresses new fuel storage, is not applicable in the PDM, and is not included in the PDTS.
6.1.1 (D6.1.1)	Because of the (1) reduced plant activities, (2) the reduced safety significance, and (3) the reduced site organization associated with the plant in the defueled condition, the responsibilities specified in this Specification for the CEO, Nuclear and the AGM,Nuclear Power Production are combined and are now the responsibility of the AGM, Nuclear. Specification 6.1.1 is rewritten to address the AGM, Nuclear responsibility to manage the facility and safely store irradiated core components at Rancho Seco while the plant is in the PDM. Attachment V contains the Rancho Seco organization which will be in effect when the PDTS are approved and implemented.
6.1.2 (D6.1.2)	The Control Room command function requirement is retained in the PDTS as Specification D6.1.2. The Shift Supervisor, who is a Certified Fuel Handler (CFH), or another qualified individual who is a CFH is responsible for the Control Room command function. The qualifications of the individual designated for the Control Room command function is modified to a CFH and addressed in the 10 CFR 55 licensed operator exemption request provided in the evaluation for Table 6.2-1 below.

Technical Specification Number	Discussion
6.2.1 (D6.2.1)	The onsite and corporate organization requirements of Specification 6.2.1 are retained in the PDTS as Specification D6.2.1. Mostly editorial changes that reflect the shutdown and defueled status of the plant are made to the requirements of this Specification. The corporate responsibility for the overall safe operation of the facility and acceptable performance of the staff is changed from the CEO, Nuclear to the General Manager. This change is consistent with the proposed organization contained in Attachment V. Reference to the training of CFHs instead of "operating staff" is made.
6.2.2 (D6.2.2)	Specifications 6.2.2 and 6.2.2a are retained in the PDTS without change and included in the PDTS as Specifications D6.2.2 and D6.2.2a, respectively. Specification 6.2.2a references Table 6.2-1. This Table is modified and retained in the PDTS as Table D6.2-1. Changes to Table 6.2-1 are addressed below in the exemption request from 10 CFR 55 Licensed Operator and Senior Licensed Operator requirements. Specification 6.2.2b is modified to reflect the necessity of maintaining at least one operator, trained and qualified commensurate with the monitoring activities and the response functions applicable in the PDM, in the Control Room when fuel is in the SFP. The Control Room operator is either a Certified Fuel Handler (CFH), trained and qualified in accordance with the CFH Training Programs (Attachment III), or a Non-Certified Operator who has been trained and qualified to stand watch in the Control Room in accordance with the Non-Certified Operator Training Programs. This modified requirement is included in the PDTS as Specification D6.2.2b. (See the evaluation for Specification 6.4.1 on page 44 of this SAR for further evaluation concerning Non-Certified Operators.) Specification 6.2.2c addresses Control Room manning requirements when fuel is in the reactor and the plant is in a mode above Cold Shutdown or Refueling; therefore, this requirement is not applicable in the PDM and is not included in the PDTS. Specification 6.2.2d is modified to reflect the need to have an individual, qualified in radiation protection practices and procedures, onsite when fuel is in the SFP. This modified requirement is included in the PDTS as Specification D6.2.2c.

Technical Specification Number	Discussion
6.2.2 (D6.2.2) (Cont.)	Specification 6.2.2e is modified to reflect the importance of requiring a trained and qualified individual to supervise all fuel handling operations. In the PDM, this individual is a CFH. Modifying this requirement to address a CFH in lieu of a NRC licensed Senior Reactor Operator is addressed in the 10 CFR 55 exemption request presented below. This modified requirement is included in the PDTS as Specification D6.2.2d. Specification 6.2.2f has been previously deleted. Specification 6.2.2g requires the existence of administrative procedures that limit overtime for facility staff who perform safety-related functions. This requirement stems from NUREG-0737, Action Item I.A.1.3, Shift Manning. In the clarification section of NUREG-0737 Action Item I.A.1.3 it states, "In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance or major plant modifications), the following overtime restrictions should be followed:..." Thus, it is clear this administrative requirement is intended for plants which operate at power and is not applicable in the PDM. Furthermore, because the plant is (1) shut down, (2) does not generate revenue, and (3) will not operate in the PDM, operating pressures no longer exist and overtime is discouraged as much as possible to minimize costs. Accidents considered credible which pose a significant risk to plant personnel and the public health and safety exist during power operation but not in the PDM. Therefore, this administrative requirement is not applicable in the PDM and is not included in the PDTS. Specification 6.2.2h is modified to reflect the replacement of NRC licensed operators with Certified Fuel Handlers. The title Shift Operations Superintendent is replaced with a general description of the position. Instead of requiring the individual who supervises the Shift Supervisors to hold a Senior Reactor Operator license pursuant to 10 CFR 55, this individual is required to be a CFH. This administrative requirement is included in the PDTS as Specification D6.2.2e.
Table 6.2-1 (Table D6.2-1)	Table 6.2-1, Shift Crew Personnel and License Requirements, is modified to reflect the shutdown status of the plant and to eliminate the requirement to maintain NRC licensed operators pursuant to 10 CFR 55 while the plant is in the PDM. Replacement of the 10 CFR 55 operator training program with the new District administered Certified Fuel Handler (CFH) Training Programs requires exemption from 10 CFR 55 requirements. The CFH Training Programs are included in the PDTS package as Attachment III to support the review and approval of the 10 CFR 55 exemption request presented below. The CFH Training Programs are provided for initial NRC review and approval. Subsequent changes to the proposed CFH Training Programs would be subject to the

Technical Specification Number	Discussion
Table 6.2-1 (Table D6.2-1) (Cont.)	requirements of 10 CFR 50.59 and are addressed in the PDTS Administrative Controls sections D6.5 and D6.8. The operator staffing levels proposed in Table D6.2-1 meet the minimum number of shift personnel specified in 10 CFR 50.54(m) for a single unit, non-operating plant. The proposed qualification and training requirements for the minimum shift crew are commensurate with the activities associated with a plant in the PDM. The following exemption request is made pursuant to 10 CFR 55.11, Specific Exemptions.

Exemption Request

The District requests an exemption from the licensed operator requirements of 10 CFR 55, Operators' Licenses while the plant is in the PDM. In place of a 10 CFR 55 licensed operator program, which is administered jointly by the District and the NRC, the District proposes to implement its own Certified Fuel Handler Training Programs. This CFH program would be initially approved by the NRC, administered by the District, subjected to 10 CFR 50.59 and PDTS Administrative Controls, and periodically inspected by the NRC.

Regulatory Requirement

10 CFR 55 establishes procedures and criteria for the issuance, modification, maintenance, and renewal of licenses to operators and senior operators who manipulate the controls of a utilization facility licensed pursuant to 10 CFR 50. These licenses are controlled and issued by the NRC.

Exemption Criteria

10 CFR 55.11 describes the criteria for exemptions from the requirements of 10 CFR 55. The Commission may grant exemptions from the requirements of the regulations of 10 CFR 55 if the Commission determines an exemption request is authorized by law and will not endanger life or property and is otherwise in the public interest.

Basis for Exemption

As a result of a public vote on June 6, 1989, the District shut down Rancho Seco Nuclear Generating Station on June 7, 1989, and completed defueling operations on December 8, 1989. The plant has been maintained in the defueled condition since December 8, 1989. On May 2, 1990, the NRC issued a Confirmatory Order which modified the facility license and prevents the District from moving fuel into the reactor building without prior NRC approval. As evaluated on pages 2 and 3 of this SAR and discussed in the NRC Safety Evaluation Report (SER) which granted an exemption from

Technical Specification Number	Discussion
Table 6.2-1 (Table D6.2-1) (Cont.)	Basis for Exemption (Continued)

simulation facility and simulator training requirements (NRC letter dated May 16, 1990, TAC No. 75520), the only credible design basis accidents remaining are loss of offsite power and a fuel handling accident. Also, because of the underwater geometric storage arrangement of the fuel assemblies, a criticality accident is not considered credible. Furthermore, the District does not intend to ever operate the Rancho Seco reactor. To formally document the District's intent to not operate Rancho Seco, the District submitted an application to the NRC on April 26, 1990 (AGM/NUC 90-114), requesting a Possession-Only License (POL). A POL would remove the District's legal authority to operate Rancho Seco as a nuclear generating station.

The District has determined the licensed operator requirement of 10 CFR 55 may be replaced with a District administered CFH program without affecting nuclear safety or endangering life or property because:

1. The reactor is defueled to the SFP,
2. Plant closure and layup activities are in progress,
3. The NRC issued a license condition which prevents the movement of fuel into the reactor building without prior NRC approval,
4. A POL is pending,
5. Minimal potential consequences are associated with accidents considered credible in the PDM, and
6. The District has no intent to operate Rancho Seco as a nuclear facility.

The CFH Training Programs provided in Attachment III for NRC review and approval are based on the guidance contained in 10 CFR 72.192 and a systems approach to training for CFH duties and responsibilities. The CFH Training Programs provide a level of operator training and certification consistent with the defueled condition of the plant and Subpart I, Training and Certification of Personnel, to 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

The change from NRC Licensed Operators to District Certified Fuel Handlers is commensurate with the change of Rancho Seco from an operating reactor facility to a permanently shutdown and completely defueled facility with all fuel stored in the Spent Fuel Pool. Furthermore, pursuant to 10 CFR 55.2(a) individuals who manipulate the "controls" of a utilization facility, licensed pursuant to 10 CFR 50, must be NRC licensed operators. "Controls" are defined in 10 CFR 55.4 as "apparatus and mechanisms the manipulation of which directly affects the reactivity or power level of the reactor." With all fuel stored in the spent fuel

Technical Specification Number	Discussion
Table 6.2-1 (Table D6.2-1) (Cont.)	Basis for Exemption (Continued)

pool, there are no controls that can be manipulated which directly or indirectly affect the reactivity or power level of the reactor. Criticality accidents are not considered credible in the PDM. Therefore, the District has concluded that the requirements of 10 CFR 55 do not apply to Rancho Seco in the PDM. The District performed a formal review of the proposed change to the training program pursuant to the requirements of 10 CFR 50.59. This review determined that the change from a 10 CFR 55 licensed operator program to a CFH program does not involve an unreviewed safety question.

As stated in the NRC's SER for the exemption previously granted from 10 CFR 55 simulator requirements,

> "The requirements of 10 CFR 55 were promulgated on the assumption that licensed operators would be controlling an operating facility which would experience transients and malfunctions from start-up through full-power operations."

Because the plant can no longer experience transients and malfunctions associated with an operating facility and will not have personnel responsible for the manipulation of controls as defined in 10 CFR 55.4, the District contends that the provisions of 10 CFR 55 are not required to protect public health and safety and that an exemption from the requirements of 10 CFR 55 to license operators or maintain Operator Licenses should be granted. The CFH Training Programs submitted are consistent with the reduced activities and the level of operator training required for a plant in the PDM.

The District continues to maintain a systems approach to training, and, consistent with that approach, has concluded that it is not necessary to provide the level of training required by 10 CFR 55 or maintain NRC-issued operator licenses at a facility in the PDM. Furthermore, the District recognizes the training elements (knowledge and skill) necessary to execute the duties and responsibilities of a CFH and has prepared programs commensurate with a plant in a defueled condition to ensure public health and safety is not compromised.

Currently, approximately 19 members of the plant staff hold a valid NRC senior operator license. By virtue of the training and qualification requirements met to obtain this license, these individuals meet the requirements for a CFH. Appropriate numbers of personnel will be trained and qualified as a CFH in accordance with the approved training programs to ensure the minimum shift crew requirements of Table D6.2-1 are met.

Technical Specification Number	Discussion
Table 6.2-1 (Table D6.2-1) (Cont.)	Basis for Exemption (Continued) Furthermore, the District agrees with the Commission's discussion in their SER that granted an exemption from a simulation facility and simulator training, and contends since criticality accidents are not considered credible in the PDM and there are no design basis accidents for the facility other than a loss of offsite power or a fuel handling accident, the need to license operators or maintain licensed operators under the provisions of 10 CFR 55 have been obviated. As stated in the NRC's SER, the training that remains is the training required to ensure personnel are qualified to perform long term monitoring and handling of spent fuel. Because the District has evaluated the Certified Fuel Handler Training Programs per the provisions of 10 CFR 50.59, determined the programs do not involve an unreviewed safety question, and developed the program contents using a systems approach to training, an exemption should be granted to avoid having a negative training impact and expending significant resources on operator training that is unnecessary for a plant in the PDM. Implementing the CFH concept in lieu of the licensed operator requirements of 10 CFR 55 would not endanger life or property and would be in the public interest and the best interest of the District ratepayers.
6.3.1 (D6.3)	Specification 6.3.1 is modified and included in the PDTS as Specification D6.3. The qualification requirement for Shift Technical Advisors (STAs) is deleted since STAs are not required in Cold Shutdown, and thus, are not required in the PDM.
6.4.1 (D6.4)	Specification 6.4.1 is modified to reference the CFH Training Programs provided for NRC review and approval, and is included in the PDTS as Specification D6.4. Retraining and replacement training of CFHs are addressed in proposed Specification D6.4 and the CFH Training Programs provided in Attachment III. The remainder of the facility staff will be trained in accordance with plant administrative procedures to maintain a level of staff competency commensurate with the activities and potential hazards associated with a plant in the PDM. Plant staff qualification requirements will continue to meet the requirements of ANSI N18.1-1971. Training requirements for the entire facility staff are addressed in the USAR; therefore, changes to these requirements are evaluated and reported to the NRC in accordance with 10 CFR 50.59. No requirement for NRC approved Non-Certified Operator or facility staff training programs is included in the PDTS. Since 10 CFR 55 does not require a licensee to have a NRC-approved training program for non-licensed operators at an operating nuclear power plant, the Non-Certified Operator Training Programs are likewise not required to be approved by the NRC while the plant is in the PDM and are not addressed in the PDTS. This conclusion is consistent with the operating plant Technical Specifications (Appendix A).

Technical Specification Number	Discussion
6.5.1, 6.5.2 (D6.5.1)	Specifications 6.5.1, Plant Review Committee (PRC), and 6.5.2, Management Safety Review Committee (MSRC), are combined, rewritten, and included in the PDTS as Specification D6.5.1, Nuclear Safety Review and Audit Committee (NSRAC). The PRC and MSRC function is combined into a single committee titled NSRAC. The majority of the responsibilities and requirements of the PRC and MSRC are retained in the new NSRAC Specification. Combining the PRC and MSRC into a single committee is commensurate with the reduced scope of activities and programs, and the review, audit, and oversight requirements necessary for a plant in the PDM. Also, combining the PRC and MSRC into the NSRAC reflects the reduced safety significance, limited credible accidents, and minimal potential consequences of credible accidents that are associated with a plant in the PDM.

The Rancho Seco working organization and plant staff are greatly reduced from that of an operating plant. It is unnecessary in the PDM to maintain the relationships between the PRC and plant operating management, the MSRC and upper plant management, and the PRC and MSRC, and these groups' independence. The proposed review, audit, and oversight requirements and management relationships for the NSRAC are commensurate with the reduced activities, scope of remaining programs, and potential accidents and consequences associated with a plant in the PDM. The facility organization which will be in effect to implement the PDTS is presented in Attachment V. The AGM, Nuclear will function as the plant manager in the PDM with the Deputy AGM, Nuclear as his qualified backup.

The function of the NSRAC will be to advise the AGM, Nuclear on the matters listed in Specification D6.5.1.1. This NSRAC function is a combination of the existing PRC and MSRC functions rewritten to reflect a plant in the PDM.

Personnel who make up the NSRAC are required to meet the current Technical Specification qualification requirements specified for the PRC or MSRC. These qualification requirements are (1) personnel filling positions in the Nuclear Organization who meet or exceed ANSI N18.1-1971, Section 4.2 or 4.4, or (2) other personnel who meet or exceed ANSI/ANS 3.1-1981, Section 4.7.2. This qualification requirement for NSRAC members is included in the PDTS as Specification D6.5.1.2. Also, the proposed number of NSRAC members (7) is commensurate with the reduced activities and site organization associated with the plant in the PDM. The Deputy AGM, Nuclear will be the NSRAC Chairman.

Technical Specification Number	Discussion
6.5.1, 6.5.2 (D6.5.1) (Cont.)	The membership, meeting frequency, quorum, review responsibility, authority, and record requirements for the NSRAC are a combination of the existing PRC and MSRC requirements rewritten to reflect a plant in the PDM. In the PDM there are no safety-related structures, systems, or components; therefore, existing MSRC review requirement 6.5.2.6h is not included in the PDTS. The NSRAC requirements are included in the PDTS subparts that comprise Specification D6.5.1. PDTS Specification D6.5.2 is not used. A Specification (D6.5.1.10) that addresses the formation of NSRAC subcommittees is included in the PDTS to provide a second level review of the NSRAC reviews required by Specification D6.5.1.7. This requirement is consistent with existing plant practice and the NRC's accepted method for screening plant changes. Specifications 6.5.1.8 and 6.5.2.8, Records, are combined, modified, and included in the PDTS as Specification D.6.5.1.11. Because review committee minutes for a specific meeting are approved at future committee meetings, NSRAC meeting minutes will be approved at the subsequent, regularly scheduled meeting. The requirement to prepare and forward NSRAC meeting minutes to the AGM, Nuclear within 14 days following a meeting is retained in the PDTS. Specification D6.5.1.11 meets the intent of the existing record requirements to provide meeting minutes to the AGM, Nuclear in a timely manner and to have the meeting minutes approved during a subsequent NSRAC meeting by the full committee.
6.5.3 (D6.5.3)	Specification 6.5.3, Technical Review and Control, is editorially modified to clarify the applicability of the review requirements for specific procedures, plans, manuals, and programs and to reflect the PDM organization. These requirements are included in the PDTS as Specification D6.5.3.
6.5.4 (D6.5.4)	The Specification 6.5.4 audit requirements applicable in the PDM are included in the PDTS unmodified. Some of these requirements are editorially modified to reflect the non-operating status and defueled condition of the plant. These audit requirements are included in the PDTS as Specification D6.5.4 with the following changes: 1. The wording contained in audit requirement 6.5.4a is editorially modified to reflect the wording in Regulatory Guide 1.33, Quality Assurance Program Requirements. This audit requirement is included in the PDTS as Specification D6.5.4a. 2. Audit requirement 6.5.4b is changed to <u>performance</u> and qualifications instead of <u>training</u> and qualifications in accordance with licensee discussions with NRC staff. The NRC desired version of this audit requirement is determined appropriate to implement in the PDM and is included in the PDTS as Specification D6.5.4b.

Technical Specification Number	Discussion
6.5.4 (D6.5.4) (Cont.)	3. The audit frequency of the REMP Manual and implementing procedures is changed to 2 years from 12 months to be consistent with the Standard RETS (NUREG-0472) and the current audit frequency specified for the ODCM and PCP. Also, the existing annual audit of REMP results is retained in the PDTS in accordance with Generic Letter 89-01. These audit requirements are included in the PDTS as Specification D6.5.4l and D6.5.4k, respectively.
	Audit reports generated from the reviews required by Specification D6.5.4 are required to be forwarded to the AGM, Nuclear, the NSRAC Chairman, and management positions responsible for the areas reviewed. This is consistent with the existing requirements.
6.6 (D6.6)	Specification 6.6, Licensee Event Report Action, is retained in the PDTS as Specification D6.6. This requirement is editorially modified to include the additional Licensee Event Report requirements contained in 10 CFR 20.405 and 73.71 and to reflect the replacement of the PRC and MSRC with the NSRAC.
6.7	This Specification describes the reporting actions required if a Safety Limit, as defined in Technical Specification Section 2, is violated. Since the Section 2 Safety Limits are not applicable in the PDM (see the evaluation for Section 2 on page 9 of this SAR), this Specification is not included in the PDTS. PDTS Section D6.7 is not used.
6.8 (D6.8)	Specification 6.8, Procedures, has been modified to reflect those procedures, plans, programs, and manuals required for a plant in the PDM and is included in the PDTS as Specification D6.8. For the PDM, the Certified Fuel Handler Training Programs and the Quality Assurance Program are added to the list of procedures, plans, manuals, and programs included in PDTS Specification D6.8.1. Specification 6.8.2 is editorially modified and included in the PDTS as Specification D6.8.2. In addition, requirements for the Radioactive Effluent Controls Program and the REMP are included in the PDTS in accordance with Generic Letter 89-01 as Specifications D6.8.3a and D6.8.3b, respectively, and Specification D6.8.4.
	Specification 6.8.1b is editorially modified to address the safe storage and handling of irradiated core components instead of refueling operations. Specification 6.8.1c is modified to reflect that the focus of surveillance and test activities conducted in the PDM is on equipment required for long term safe storage of irradiated core components rather than on safety-related equipment. These requirements are included in the PDTS as Specifications D6.8.1b and D6.8.1c, respectively.

Technical Specification Number	Discussion
6.8 (D6.8) (Cont.)	As previously justified in Proposed Technical Specification Amendment No. 155 and approved by the NRC in Rancho Seco License Amendment No. 98, the District possesses a deviation from the Standard RETS requirement relating to the dose projection limit for operation of the liquid effluent radwaste treatment system. The NRC approved a 31-day dose projection limit of 8 1/3 percent of the annual 10 CFR 50, Appendix I dose guidelines on liquid effluent radwaste treatment system operation, instead of the Standard RETS 31-day dose project limit of 2 percent of the annual guidelines. This relief is necessary because Rancho Seco is a dry site, thus making compliance with 10 CFR 50, Appendix I more difficult. Therefore, the 31-day dose projection limit which currently exists in the Rancho Seco Technical Specifications is retained for the PDM and is included in the PDTS as Specification D6.8.3a.7. Also, due to the absence of Iodine-131 and Iodine-133 resulting from the extended plant shutdown and the absence of any production mechanism for these isotopes in the PDM, requirements addressing these isotopes are excluded from PDTS Specification D6.8.3a.10. This program requirement for radionuclides in particulate form gives reasonable assurance that the release of gaseous effluent in the PDM will be maintained within the 10 CFR 50, Appendix I guidelines.
6.9.1 (D6.9.1)	Specifications 6.9.1.1 through 6.9.1.4 provide requirements for submitting various startup related reports. These reporting requirements are not applicable in the PDM and are not included in the PDTS. A new Specification that addresses the submittal of routine reports specified in the PDTS is included in the PDTS as Specification D6.9.1.
6.9.2 (D6.9.2, D6.9.3)	Specification 6.9.2, Radiological Reports, is editorially modified to conform with the format of the Standard Technical Specifications and is included in the PDTS as Specification D6.9.2, Annual Radiological Reports. The reports included in D6.9.2 are: 1. D6.9.2.1, Annual Occupational Radiation Exposure Report, 2. D6.9.2.2, Annual Exposure Report, and 3. D6.9.2.3, Annual Radiological Environmental Operating Report. Specification 6.9.2.2, Annual Radiological Environmental Operating Report, was modified to meet the requirements of Generic Letter 89-01 and is included in the PDTS as Specification D6.9.2.3. Specification 6.9.2.3, Semiannual Radioactive Effluent Release Report, was modified to meet the requirements of Generic Letter 89-01 and is included in the PDTS as Specification D6.9.3.

Technical Specification Number	Discussion
6.9.3 (D6.9.4)	This Specification details the requirements for the content and submittal of the Monthly Report. This report addresses primarily plant operating statistics and experiences. This reporting requirement is based on the reporting requirement contained in 10 CFR 50.59(b)(2). Because activities are greatly reduced for a plant in the PDM, the reporting period is changed to annually. This is consistent with the requirements of 10 CFR 50.59(b)(2). The required content of the report is editorially modified to reflect the significant reduction in and different type of activities anticipated for a plant in the PDM. The Annual Report is included in the PDTS as Specification D6.9.4.
6.9.4 (D6.9.5)	This Specification provides requirements on reportability pursuant to 10 CFR 20.405, 50.73, and 73.71. The details of 10 CFR 50.73 have been removed and replaced with a reference to meet 10 CFR 50.73 to avoid a conflict with potential future changes to this code. This Specification is editorially modified to include the additional Licensee Event Report requirements contained in 10 CFR 20.405 and 73.71. These reportability requirements are included in the PDTS as Specification D6.9.5.
6.9.5 (D6.9.7)	The Special Reports required by this Specification are not included in the PDTS for the following reasons: 6.9.5.A was a one time only report required for 1977. 6.9.5.B,C,D,F, and P are associated with a plant with an operating reactor. 6.9.5.E was previously relocated to the Fire Protection Plan pursuant to Generic Letter 88-12 and Rancho Seco License Amendment No. 115. 6.9.5.G,H,I,J,K,M,N, and O have been removed and relocated per Generic Letter 89-01. 6.9.5.L was previously deleted. A general statement regarding the submittal of Special Reports is included in the PDTS as Specification D6.9.7. This general statement is in conformance with the Standard Technical Specifications.
6.9.6 (D6.9.6)	This Specification, Environmental Reports, is retained in the PDTS as Specification D6.9.6. Retaining these reporting requirements contributes to the justification for deleting Appendix B from the Rancho Seco Facility Operating License. An editorial clarification is made to Specification 6.9.6.2, and this Specification is included in the PDTS as Specification D6.9.6b.

Technical Specification Number	Discussion
6.10 (D6.10)	The requirements for records retention are included in the PDTS as Specification D6.10. The District proposes changes to two existing record retention requirements (6.10.2l and 6.10.2n). The remaining existing record retention requirements are retained in the PDTS. At the time the Facility Operating License is modified to a Possession-Only License, the record retention requirements will be reviewed and modified as appropriate. Some record retention requirements are added to address a plant in the PDM. Changes and additions made to this Specification are described below. D6.10.1j is added to ensure records and logs of the operation of plant systems and equipment in the PDM are retained for at least five years. This requirement is consistent with existing Specification 6.10.1a (D6.10.1a). Specification 6.10.2l, Environmental Qualification Records, is modified to ensure Environmental Qualification (EQ) records previously generated under the provisions of Appendix A Specification 6.14 are retained for the duration of the Facility Operating License. This requirement is included in the PDTS as Specification D6.10.2l. An exemption from 10 CFR 50.49, Environmental Qualification, and justification for the exclusion of Specification 6.14 from the PDTS are presented below in the evaluation for Specification 6.14. Specification 6.10.2n, Snubber Service Life Records, is modified to ensure the snubber records previously generated under the provisions of this Specification are retained for the duration of the Facility Operating License. This requirement is included in the PDTS as Specification D6.10.2n. A requirement for the retention of records performed for reviews of changes made to the ODCM, REMP MANUAL, and the PCP is added to the PDTS as Specification D6.10.2o in accordance with Generic Letter 89-01. A new requirement for the retention of records related to safe and effective Decommissioning of the facility is included in the PDTS as Specification D6.10.2p in accordance with 10 CFR 50.75(g). Specification D6.10.2q is added to ensure records of NSRAC meetings are retained.
6.11 (D6.11)	This Specification is included in the PDTS as Specification D6.11. Reference to 10 CFR 19 is added to this Specification for completeness.

Technical Specification Number	Discussion
6.12	This Specification was previously deleted.
6.13 (D6.12)	The requirements for access control to High Radiation Areas is retained in the PDTS as Specification D6.12.
6.14	Specification 6.14, Environmental Qualification (EQ), is not included in the PDTS since an accident that results in a harsh environment which could affect the ability of electrical equipment to maintain SFP cooling, water level, or water temperature, or could result in potential offsite exposures comparable to 10 CFR 100 limits, is not credible in the PDM. But, because EQ requirements are addressed in 10 CFR 50.49, the following exemption request is made in accordance with 10 CFR 50.12, Specific Exemptions.

Exemption Request

The District requests an exemption from the provisions of 10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants.

Regulatory Requirements

10 CFR 50.49 requires the District to maintain a comprehensive program for qualifying electrical equipment important to safety, including updated records. Equipment covered by 10 CFR 50.49 includes both safety-related and non safety-related electrical equipment that is relied upon to remain functional during and following design basis events to ensure:

1. The integrity of the Reactor Coolant System,

2. The capability to shut down the reactor and maintain it in a safe shutdown condition, and

3. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to 10 CFR 100 limits.

Exemption Criteria

10 CFR 50.12 describes the criteria for exemptions from the requirements of 10 CFR 50. The NRC may grant exemptions from Part 50 regulations if special circumstances are present and an undue risk will not be presented to the public health and safety. In the discussion below, the District will address the following exemption criteria of 10 CFR 50.12(a)(2):

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule;

Technical Specification Number	Discussion
6.14 (Cont.)	Exemption Criteria (Continued)

(iii) Compliance would result in undue hardship on other costs that are significantly in excess of those contemplated when the rule was adopted....

Basis for Exemption

Rancho Seco was permanently shut down on June 7, 1989, and completely defueled on December 8, 1989. Also, Confirmatory Order dated May 2, 1990, prevents the District from moving fuel into the Reactor Building without prior NRC approval; therefore, the spectrum of postulated accidents considered credible and the resulting potential consequences are much less than those considered when the NRC promulgated the EQ rule.

As stated on page 2 of this SAR and discussed in the granting of the partial exemption to 10 CFR 55 (NRC to SMUD letter dated May 16, 1990, TAC No. 75520), the design basis accidents considered credible in the PDM are reduced to a Loss of Offsite Power (LOOP) and a Fuel Handling Accident. These design basis accidents do not create a harsh environment as defined in 10 CFR 50.49. Plant energy sources capable of producing a harsh environment have been de-energized, except for the Auxiliary Steam System (used for heating and liquid radwaste processing). A District evaluation concluded that a line break in the Auxiliary Steam System cannot adversely affect the ability to maintain SFP cooling, water level, or water temperature. Furthermore, another District evaluation concluded that the design basis accidents considered credible in the PDM result in offsite exposures much less than 10 CFR 100 limits. (See pages 2 and 3 of this SAR.)

The three criteria specified in the Regulatory Requirements section of this exemption request, which define the scope of the equipment required to be maintained as EQ equipment per 10 CFR 50.49, are not applicable in the PDM because (1) the Reactor Coolant System is not required to be operable and is de-energized, (2) the reactor is shut down and void of fuel, and (3) no accidents whose consequences could approach the 10 CFR 100 limits are considered credible. Since no electrical equipment is required to be maintained as EQ equipment in the PDM, an EQ Specification is not required to be included in the PDTS.

The requirements of 10 CFR 50.49 were predicated upon the need to ensure the spectrum of postulated accidents from power operations would not endanger public health and safety by exceeding the requirements of 10 CFR 100. Because Rancho Seco is not capable of an accident comparable to 10 CFR 100 limits and not capable of creating a harsh environment per 10 CFR 50.49, compliance with 10 CFR 50.49 would not serve the underlying purpose of the rule

Technical Specification Number	Discussion
6.14 (Cont.)	Basis for Exemption (Cont.) when the rule was promulgated (10 CFR 50.12(a)(2)(ii)). Also, expending resources to maintain an EQ program in the PDM, including updated records, would result in undue hardship and costs for a permanently shutdown and completely defueled reactor that are significantly in excess of those contemplated when the rule was adopted (10 CFR 50.12(a)(2)(iii)).
6.15 (D6.13)	This Specification, PCP, is revised in accordance with the guidelines of Generic Letter 89-01, and is included in the PDTS as Specification D6.13.
6.16 (D6.14)	This Specification, ODCM and REMP, is revised in accordance with the guidelines of Generic Letter 89-01, and is included in the PDTS as Specification D6.14.
6.17	This Specification, Major Changes to Radioactive Waste Treatment Systems, is relocated to the PCP in accordance with the guidelines of Generic Letter 89-01. Major changes to radioactive waste treatment systems will be reviewed by the NSRAC as required by PDTS Specification D6.5.1.7i.
6.18	This Specification, Postaccident Sampling (PAS), is not applicable in the PDM because the PAS system is designed to monitor reactor accidents; thus this Specification is not included in the PDTS.

The requirements of Appendix B to Operating License No. DPR-54 have been previously justified for deletion from the Technical Specifications in Proposed Amendment No. 102. The administrative controls that were required to be retained in Appendix A so that the Appendix B Specifications could be deleted are now also contained within the proposed PDTS (Appendix C). Sufficient justification and administrative controls exist in this submittal and previous submittals (PA-102) to conclude that the Appendix B Specifications are adequately addressed in the PDTS and the Appendix A Technical Specifications such that Appendix B should be deleted as part of the approval of the Appendix C Technical Specifications. Attachment VII contains a copy of the most recent submittal of Proposed Amendment No. 102. Also, Attachment VII contains a version of this proposed amendment that addresses the characteristics of Appendix C that also justify the deletion of Appendix B. Based upon the content of both Appendix A and Appendix C, the Appendix B Technical Specifications should be deleted independent of the approval of Appendix C.

Safety Analysis Summary

The Rancho Seco PDTS provide the controls necessary for the protection of the health and safety of the public in the PDM. These PDTS are intended to act as a stand alone document and to take the place of Appendices A and B to Operating License No. DPR-54 during the current extended outage. A new mode, the Permanently Defueled Mode (PDM), is defined in the PDTS to describe the plant condition under which the Appendix C Technical Specifications are applicable. The Appendix A Technical Specifications would become applicable if the Commission allowed the District to move fuel back into the Reactor Building.

The PDTS were developed based on an evaluation of credible accidents applicable in the PDM, and the equipment needed to mitigate these accidents. Two accidents evaluated in Chapter 14 of the USAR are considered credible in the PDM: 1) a fuel handling accident and 2) a LOOP. The LOOP accident is determined to have no adverse impact on the health and safety of the public in the PDM, while the dropped fuel assembly accident results in very small calculated exposures to the maximum exposed individual.

The existing Rancho Seco Technical Specifications (Appendices A and B) were evaluated and the appropriate requirements incorporated into the PDTS and other plant documents (i.e., ODCM, REMP Manual, and PCP) to ensure that plant closure activities will result in radioactive releases from the plant which are as low as is reasonably achievable. A detailed description of this evaluation is provided in this safety analysis.

The PDTS presented in Proposed Amendment No. 182, Revision 2 provide technical and administrative controls sufficient to assure the protection of the health and safety of the public. For the plant in the PDM, the PDTS (Appendix C) maintain the margin of safety previously provided by the Appendix A and B Technical Specifications when the plant was operating.

NO SIGNIFICANT HAZARDS CONSIDERATION

The District has reviewed the proposed PDTS against each of the criterion of 10 CFR 50.92 and concludes that the proposed changes as described and evaluated in the above safety analysis do not:

a. Involve a significant increase in the probability or consequences of an accident previously evaluated. There are only two credible accidents in the Permanently Defueled Mode, a fuel handling accident and a LOOP. The changes proposed do not increase the probability of either of these accidents since the LOOP is not controllable by the plant, and the requirements for testing of the fuel handling bridge remain unchanged. The consequences of the two credible accidents are bounded by the previous analyses for these accidents. The fuel handling accident scenario remains unchanged, but the consequences of this accident are significantly reduced due to the length of the decay time of the fuel. The extended time period available to restore offsite power before only 6.75 feet of water boils from the SFP (a minimum of 11.86 days as of June 7, 1990) provides ample time to take corrective action to ensure fuel cladding damage does not occur following a loss of offsite power.

b. Create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed addition to the Operating License, the PDTS (Appendix C), applies only when the reactor is in the PDM. Therefore, only those activities and potential accidents associated with the SFP need be considered. Because the ability to maintain an adequate shutdown margin in the SFP is not affected, no physical changes to the SFP or SFC are being made, and systems required to safely store the spent fuel in the PDM will continue to be maintained, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

c. Involve a significant reduction in a margin of safety since the margin of safety for the fuel handling accident is unchanged, while the margin of safety for the LOOP event is not significantly reduced given the extended time period available to restore offsite power or to find an alternate means of adding water to the SFP.

Based on the evaluation provided above, the District has concluded that the proposed PDTS do not constitute a significant hazard to the public and do not endanger the public's health and safety.

TABLE 1

DECAY HEAT LOAD AND SFP HEAT UP RATE VS. TIME

DATE	SPENT FUEL DECAY HEAT (MILLION BTU/HR)	TIME TO REACH 212°F FROM 120°F IN THE SPENT FUEL POOL (HOURS)	(DAYS)	TIME TO REACH 212°F FROM 120°F AND VAPORIZE 6.75 FT. OF SPENT FUEL POOL WATER (HOURS)	(DAYS)
1/3/90	3.600	74.4	3.1	220.25	9.18
6/7/90	2.786	96.12	4.01	284.56	11.86
6/7/91	1.862	143.85	5.99	425.86	17.74
6/7/94	1.372	195.21	8.13	577.88	24.08
6/7/99	1.182	230.97	9.62	683.76	28.49
6/7/09	0.897	298.73	12.45	884.33	36.85

TABLE 2

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

PUMPS

PUMP	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
P-108A	TDI DIESEL FO PUMP "A"	YES	NO
P-108B	TDI DIESEL FO PUMP "B"	YES	NO
P-108C	TDI DIESEL FO PUMP "A"	YES	NO
P-108D	TDI DIESEL FO PUMP "B"	YES	NO
P-236	MAKEUP	YES	NO
P-238A	HP INJECTION	YES	NO
P-238B	HP INJECTION	YES	NO
P-261A	DECAY HEAT REMOVAL	YES	NO
P-261B	DECAY HEAT REMOVAL	YES	NO
P-291A	REACTOR BUILDING SPRAY	YES	NO
P-291B	REACTOR BUILDING SPRAY	YES	NO
P-318	DUAL DR AUXILIARY FEEDWATER	YES	NO
P-319	MOTOR-DRIVEN AUX FEEDWATER	YES	NO
P-472A	NUCLEAR SERVICE RAW WATER	YES	NO
P-472B	NUCLEAR SERVICE RAW WATER	YES	NO
P-482A	NUCLEAR SERVICE COOLING WATER	YES	NO
P-482B	NUCLEAR SERVICE COOLING WATER	YES	NO
P-705A	BORIC ACID	YES	NO
P-705B	BORIC ACID	YES	NO
P-888A	BRUCE GM DIESEL FO PUMP "A"	YES	NO
P-888B	BRUCE GM DIESEL FO PUMP "A"	YES	NO
P-888C	BRUCE GM DIESEL FO PUMP "B"	YES	NO
P-888D	BRUCE GM DIESEL FO PUMP "B"	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: REACTOR COOLANT

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
RIVVWX	Rx Internals Vent	YES	NO
RIVVWZ	Rx Internals Vent	YES	NO
RIVVXW	Rx Internals Vent	YES	NO
RIVVXY	Rx Internals Vent	YES	NO
RIVVYX	Rx Internals Vent	YES	NO
RIVVYZ	Rx Internals Vent	YES	NO
RIVVZW	Rx Internals Vent	YES	NO
RIVVZY	Rx Internals Vent	YES	NO
HV-21505	PORV Blocking	YES	NO
HV-21522	Press Hi Pt Vent-Otbd	YES	NO
HV-21528	Press Hi Pt Vent-Inbd	YES	NO
PSV-21506	Pressurizer Safety	YES	NO
PSV-21507	Pressurizer Safety	YES	NO
PSV-21511	Power Oper Relief Valve	YES	NO
HV-20533	Stm Gen E-205A INBD Hi Pt Vent	YES	NO
HV-20534	Stm Gen E-205B INBD Hi Pt Vent	YES	NO
HV-20579	Stm Gen E-205A OTBD Hi Pt Vent	YES	NO
HV-20580	Stm Gen E-205B OTBD Hi Pt Vent	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: MAKE-UP AND PURIFICATION

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
SFV-22005	Letdown to E-220A	YES	NO
SFV-22006	Letdown to E-220B and E-220C	YES	NO
SFV-22009	Letdown-Otbd Isol	YES	NO
SFV-22023	Letdown-Inbd Isol	YES	NO
SFV-22025	Letdown from Stm Gen 205A	YES	NO
SFV-23645	HP INJ/MU Pump Min Flow	YES	NO
SFV-23646	HP INJ/MU Pump Min Flow	YES	NO
SFV-24004	RCP Seal Water Ret Inbd	YES	NO
SFV-24013	RCP Seal Water Ret Otbd	YES	NO
BWS-019	Supply from CBAT	NO	NO
BWS-044	Supply from CBAT	NO	NO
HV-23801	Disch to Rx Inlet Loop B	YES	NO
HV-23802	Disch to Aux Spray	YES	NO
SFV-23508	Disch from Make-up Tank	YES	NO
SFV-23604	Makeup Line Isol	YES	NO
SFV-23616	RCP Seal Supply	YES	NO
SFV-23809	Disch to Rx Inlet Loop A	YES	NO
SFV-23810	Disch to Rx Inlet Loop B	YES	NO
SFV-23811	Disch to Rx Inlet Loop A	YES	NO
SFV-23812	Disch to Rx Inlet Loop B	YES	NO
SIM-002	Make-up Pump Disch	YES	NO
SIM-019	RCP 210A Seal Sup	YES	NO
SIM-020	RCP 210B Seal Sup	YES	NO
SIM-021	RCP 210C Seal Sup	YES	NO
SIM-022	RCP 210D Seal Sup	YES	NO
SIM-023	RCP "A" Seal Inj Vent	YES	NO
SIM-025	RCP "C" Seal Inj Vent	YES	NO
SIM-026	RCP "D" Seal Inj Vent	YES	NO
SIM-036	Disc to Rx Inlet Loop A	YES	NO
SIM-037	Disc to Rx Inlet Lp A	YES	NO
SIM-040	Disc to Rx Inlet Lp A	YES	NO
SIM-041	Disc to Rx Inlet Lp A	YES	NO
SIM-043	Disch from BST	YES	NO
SIM-045	HP Inj Pump 238B Disch	YES	NO
SIM-047	Disch to Rx Inlet Lp B	YES	NO
SIM-049	Disch to Rx Inlet Lp B	YES	NO
SIM-050	Disch to Rx Inlet Lp B	YES	NO
SIM-052	Disch from BST	YES	NO
SIM-058	HP Inj Pump 238A Disch	YES	NO
SIM-078	HP Inj Pump "B" Min Flow	YES	NO
SIM-079	Makeup Pump Min Flow	YES	NO
SIM-081	HP Inj Pump "A" Min Flow	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: DECAY HEAT REMOVAL

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
BWS-003	Disch Fr BWST to P-261B	YES	NO
BWS-004	Disch Fr BWST to P-261A	YES	NO
BWS-045	Boric Ac Pump Disc To DHR	NO	NO
CBS-035	Spray Add to P-261B	NO	NO
CBS-036	Spray Add to P-261A	NO	NO
CFS-001	Inj from Core Flood Tk 265A	YES	NO
CFS-002	Inj from Core Flood Tk 265B	YES	NO
CFS-003	CFT Sample Isolation	YES	NO
CFS-004	CFT Drain Line Isol	YES	NO
CFS-005	N2 Supply to CFT 265A	YES	NO
CFS-006	N2 Supply to CFT 265B	YES	NO
CFS-009	CFT 265A Fill Line	YES	NO
CFS-010	CFT 265B Fill Line	YES	NO
CFS-011	CFT 265A Fill Fr BAAP	YES	NO
CFS-012	CFT 265B Fill Fr BAAP	YES	NO
DHS-003	P-261-B Suct Fr BWST	YES	NO
DHS-004	P-261-B Suct Fr BWST	YES	NO
DHS-007	Decay Ht Rem Pmp 261A Disch	YES	NO
DHS-008	Decay Ht Rem Pmp 261B Disch	YES	NO
DHS-015	Decay Ht Ret from E-260A	YES	NO
DHS-016	Decay Ht Ret from E-260B	YES	NO
DHS-017	Dec Ht Rem to CFT "A"	YES	NO
DHS-018	Dec Ht Rem to CFT "B"	YES	NO
DHS-038	Fuel Xfer Canal Fill	YES	NO
DHS-039	Fuel Xfer Canal Fill	YES	NO
DHS-059	Dec Ht Rem to Pres Spr	YES	NO
DHS-497	Fuel Xfer Canal Fill	YES	NO
DHS-498	Fuel Xfer Canal Fill	YES	NO
HV-20001	Rx Cool Supply to DH Rem Pmps	YES	NO
HV-20002	Rx Cool Supply to DH Rem Pmps	YES	NO
HV-20003	DH Remove Dump to Sump	YES	NO
HV-20005	Dec Ht Supply to P-261B	YES	NO
HV-20006	Dec Ht Supply to P-261A	YES	NO
HV-26007	Dec Ht Disch to P-238A	YES	NO
HV-26008	Dec Ht Disch to P-238B	YES	NO
HV-26011	Dec Ht Rem To Press Sp	YES	NO
HV-26046	Dec Ht Rem X Conn to SFP	YES	NO
HV-26047	Dec Ht Rem X Conn to SFP	YES	NO
HV-26105	P-261 "A" Suct fr Sump	YES	NO
HV-26106	P-261 "B" Suct fr Sump	YES	NO
HV-26515	CFT "A" Drain/Samp	YES	NO
HV-26516	CFT "B" Drain/Samp	YES	NO
HV-26517	CFT "A" N2 Sup/Fill	YES	NO
HV-26518	CFT "B" N2 Sup/Fill	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: DECAY HEAT REMOVAL (Continued)

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
PSV-20004	Dec Ht Rem Ret Rel	NO	NO
PSV-25009	P-261A Suct fr BWST Relief	NO	NO
PSV-25010	P-261B Suct fr BWST Relief	NO	NO
PSV-26101	P-261A Disch Relief	NO	NO
PSV-26102	P-261B Disch Relief	NO	NO
PSV-26109	P-261A Suct fr Sump Relief	NO	NO
PSV-26110	P-261B Suct fr Sump Relief	NO	NO
PSV-26509	Core Flood Tank 265A Safety	NO	NO
PSV-26510	Core Flood Tank 265B Safety	NO	NO
RCS-001	Dec Ht/Core Flood to Rx Vess	YES	NO
RCS-002	Dec Ht/Core Flood to Rx Vess	YES	NO
RCS-042	N2 to Nitro Prehtr	YES	NO
SFV-25003	P-260A Suct fr BWST	YES	NO
SFV-25004	P-260B Suct fr BWST	YES	NO
SFV-26005	E-260A Disch to RCS	YES	NO
SFV-26006	E-260B Disch to RCS	YES	NO
SFV-26039	E-260A Disch	YES	NO
SFV-26040	E-260B Disch	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50 55a IN-SERVICE TESTING PROGRAM

SYSTEM: REACTOR BUILDING SPRAY

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
CBS-001	R B S r P 291A Suct	YES	NO
CBS-002	R B Spr P 291B Suct	YES	NO
CBS-005	R B Spr P 291A Disc	YES	NO
CBS-006	R B Spr P 291B Disc	YES	NO
CBS-009	R B Spr A-Inbd Isol	YES	NO
CBS-010	R B Spr B-Inbd Isol	YES	NO
CBS-021	Sp Add Tank 290A Disc	YES	NO
CBS-022	Sp Add Tank 290B Disc	YES	NO
CBS-027	Sp Add Tank 290B Disc	YES	NO
CBS-028	Sp Add Tank 290A Disc	YES	NO
CBS-029	NAOH to Dec Ht Rem	YES	NO
CBS-030	NAOH to Dec Ht Rem	YES	NO
CBS-033	NAOH to Dec Ht Rem	YES	NO
CBS-034	NAOH to Dec Ht Rem	YES	NO
CBS-504	Rx Bldg Sp Add Tank 290B Vac Bkr	YES	NO
CBS-505	Rx Bldg Sp Add Tank 29[illegible] Vac Bkr	YES	NO
PSV-29117	Cont Sp Disch Rel	NO	NO
PSV-29118	Cont Sp Disch Rel	NO	NO
SFV-29015	Sup to Eject 292A	YES	NO
SFV-29016	Sup to Eject 292B	YES	NO
SFV-29107	Rx Bldg Spray "A" Otbd Isol	YES	NO
SFV-29108	Rx Bldg Spray "B" Otbd Isol	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: HP AND AUXILIARY TURBINES

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
TV-1	Main Turbine Throttle	YES	NO
TV-2	Main Turbine Throttle	YES	NO
TV-3	Main Turbine Throttle	YES	NO
TV-4	Main Turbine Throttle	YES	NO
HV-20560	Stm to Aux Steam	NO	NO
HV-20565	Stm to Aux Steam	NO	NO
HV-20597	Stm to Rhtrs A and C	NO	NO
HV-20598	Stm to Rhtrs B and D	NO	NO
HV-32243	Stm Drain and Pegging Steam	NO	NO
HV-35069	Steam Drain	NO	NO
HV-35070	Steam Drain	NO	NO
HV-20517	Atmos Stm Dump Iso	NO	NO
HV-20518	Atmos Stm Dump Iso	NO	NO
HV-20521	Turb BP Iso "A"	NO	NO
HV-20522	Turb BP Iso "B"	NO	NO
HV-20569	Supply to Aux FP Turbine	NO	NO
HV-20570	Stm drain to LP Cond	NO	NO
HV-20571	Stm drain to LP Cond	NO	NO
HV-20596	Supply to Aux FP Turbine	NO	NO
MSS-051	Supply to Aux FP	YES	NO
MSS-052	Supply to Aux FP	YES	NO
PSV-20533	Main Steam Relief	YES	NO
PSV-20534	Main Steam Relief	YES	NO
PSV-20544	Main Steam Relief	YES	NO
PSV-20545	Main Steam Relief	YES	NO
PSV-20546	Main Steam Relief	YES	NO
PSV-20547	Main Steam Relief	YES	NO
PSV-20548	Main Steam Relief	YES	NO
PSV-20549	Main Steam Relief	YES	NO
PSV-20550	Main Steam Relief	YES	NO
PSV-20551	Main Steam Relief	YES	NO
PSV-20552	Main Steam Relief	YES	NO
PSV-20553	Main Steam Relief	YES	NO
PSV-20554	Main Steam Relief	YES	NO
PSV-20555	Main Steam Relief	YES	NO
PSV-20556	Main Steam Relief	YES	NO
PSV-20557	Main Steam Relief	YES	NO
PSV-20558	Main Steam Relief	YES	NO
PSV-20559	Main Steam Relief	YES	NO
PV-20561	Main Turbine Bypass	NO	NO
PV-20562A	Atmos Steam Dump	NO	NO
PV-20562B	Atmos Steam Dump	NO	NO
PV-20562C	Atmos Steam Dump	NO	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: HP AND AUXILIARY TURBINES (Continued)

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
PV-20563	Main Turbine Bypass	NO	NO
PV-20564	Main Turbine Bypass	NO	NO
PV-20566	Main Turbine Bypass	NO	NO
PV-20571A	Atmos Steam Dump	NO	NO
PV-20571B	Atmos Steam Dump	NO	NO
PV-20571C	Atmos Steam Dump	NO	NO
MSS-068	Aux FP Turbine Exh	YES	NO
HV-30801	Aux FP Turbine Stop	YES	NO

SYSTEM: STEAM GENERATOR

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
FV-20525	Main Feed to SG 205A	NO	NO
FV-20526	Main Feed to SG 205B	NO	NO
FV-20527	Emerg FW Stop to SG 205A	YES	NO
FV-20528	Emerg FW Stop to SG 205B	YES	NO
FV-20531	Emerg Feed to SG205A	YES	NO
FV-20532	Emerg Feed to SG205B	YES	NO
FV-20575	Feed BP to SG 205A	NO	NO
FV-20576	Feed GP to SG 205B	NO	NO
FWS-061	Emerg Feed to SG 205A	YES	NO
FWS-062	Emerg Feed to SG 205B	YES	NO
FWS-101	Stm Gen Cleaning	YES	NO
HV-20515	Main Feed to SG 205B	YES	NO
HV-20516	Main Feed to SG 205A	YES	NO
HV-20529	Main Feed Stop to SG 205A	NO	NO
HV-20530	Main Feed Stop to SG 205B	NO	NO
HV-20577	Emerg Feed to SG 205A	YES	NO
HV-20578	Emerg Feed to SG 205B	YES	NO
HV-20581	Emerg Feed to SG 205A	YES	NO
HV-20582	Emerg Feed to SG 205B	YES	NO
HV-20611	SG Dr Boost Pump Disc	YES	NO
HV-31826	Aux FP Crossover	YES	NO
HV-31827	Aux FP Crossover	YES	NO
FWS-102	Stm Gen Cleaning Iso	YES	NO
HV-20609	Stm Gen 205A Drain	YES	NO
FWS-080	Stm Gen Booster Pump Disch	YES	NO
HV-20610	Stm Gen 205B Drain	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: HIGH PRESSURE FEEDWATER HEATER

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
FWS-047	Aux FP 318 Disch	YES	NO
FWS-048	Aux FP 319 Disch	YES	NO
FWS-049	Aux FP 318 Min Flow	YES	NO
FWS-050	Aux FP 319 Min Flow	YES	NO

SYSTEM: CONDENSER

MCM-059	CST Disch to Aux FP P-318	YES	NO
MCM-060	CST Disch to Aux FP P-319	YES	NO
PSV-31800	Aux FP P-318 Suct Rel	NO	NO
PSV-31900	Aux FP P-319 Suct Rel	NO	NO
PSV-35804	CST Relief/Vac Bkr	NO	NO
PSV-35805	CST Relief/Vac Bkr	NO	NO

SYSTEM: AUXILIARY STEAM

ASC-048	Aux Steam to RB-Otbd	YES	NO
ASC-049	Aux Steam to RB-Inbd	YES	NO

SYSTEM: COMPONENT COOLING WATER

CCW-036	CCW Supply to RB-INBD	YES	NO
SFV-46014	CCW Supply to RB-OTBD	YES	NO
SFV-46203	CCW Ret FR RB - INBD	YES	NO
SFV-46204	CCW Ret FR RB - OTBD	YES	NO
CCW-194	CCW Supply to CRD-INBD	YES	NO
SFV-46906	CCW Supply to CRD-OTBD	YES	NO
SFV-46907	CCW Ret FR CRD-INBD	YES	NO
SFV-46908	CCW Ret FR CRD-OTBD	YES	NO

SYSTEM: NUCLEAR SERVICE RAW WATER

PSV-47205A	Pump 472A Vac Bkr	NO	NO
PSV-47205B	Pump 472B Vac Bkr	NO	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: NUCLEAR SERVICE COOLING WATER

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
SFV-26016	Dec HT Rem Clr 260B Inlet	YES	NO
SFV-26017	Dec HT Rem Clr 260A Inlet	YES	NO
SFV-26018	Dec HT Rem Clr 260B Outlet	YES	NO
SFV-26019	Dec HT Rem Clr 260A Outlet	YES	NO
PSV-48411	Nuc Svc Clg Wtr Surge Tank	NO	NO
PSV-48412	Nuc Svc Clg Wtr Surge Tank	NO	NO
PSV-48415	Nuc Svc Clg Wtr Surge Tank Vac Bkr	NO	NO
PSV-48416	Nuc Svc Clg Wtr Surge Tank Vac Bkr	NO	NO
SFV-50005	Rx Bldg Clg Unit 500A Inlet	YES	NO
SFV-50006	Rx Bldg Clg Unit 500B Inlet	YES	NO
SFV-50007	Rx Bldg Clg Unit 500C Inlet	YES	NO
SFV-50008	Rx Bldg Clg Unit 500D Inlet	YES	NO
SFV-50009	Rx Bldg Clg Unit 500A Outlet	YES	NO
SFV-50010	Rx Bldg Clg Unit 500B Outlet	YES	NO
SFV-50011	Rx Bldg Clg Unit 500C Outlet	YES	NO
SFV-50012	Rx Bldg Clg Unit 500D Outlet	YES	NO

SYSTEM: TDI DIESEL FUEL OIL

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
DFO-318	Diesel FO Pump 108B Disch	YES	NO
DFO-319	Diesel FO Pump 108A Disch	YES	NO
DFO-322	Diesel FO Pump 108D Disch	YES	NO
DFO-323	Diesel FO Pump 108C Disch	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: REACTOR BUILDING HEATING, VENTILATING AND AIR COND

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
HGS-005	H2 Purge Intake	YES	NO
HGS-010	H2 Purge Intake	YES	NO
HGS-012	ILRT Test Conn	YES	NO
HGS-013	ILRT Test Conn	YES	NO
HV-53617	H2 Purge Inbd Iso	YES	NO
HV-53618	H2 Purge Inbd Iso	YES	NO
HV-53620	H2 Recombiner	YES	NO
HV-536[illegible]	H2 Recombiner	YES	NO
HV-53622	H2 Recombiner	YES	NO
HV-53623	H2 Recombiner	YES	NO
HV-70040	H2 Mon Return	YES	NO
HV-70041	RB Air Sample	YES	NO
HV-70042	RB Air Sample	YES	NO
HV-70043	Pass Return	YES	NO
HV-70044	Pass Return	YES	NO
HV-70045	Pass Atmos Sample	YES	NO
HV-70046	Pass Atmos Sample	YES	NO
HV-70047	H2 Mon Return	YES	NO
SFV-53503	RB Purge Inlet-Otbd	YES	NO
SFV-53504	RB Purge Inlet-Inbd	YES	NO
SFV-53603	RB Purge Inlet-Inbd	YES	NO
SFV-53604	RB Purge Outlet-Inbd	YES	NO
SFV-53605	RB Purge Outlet-Inbd	YES	NO
SFV-53610	RB Purge Inlet-Inbd	YES	NO
SFV-53612	From H2 Purge Ex Blr "B"	YES	NO
SFV-53613	From H2 Purge Ex Blr "A"	YES	NO
SFV-53615	H2 Purge Exh Isol	YES	NO
SFV-53616	H2 Purge Exh Isol	YES	NO

SYSTEM: COOLANT RADWASTE

SFV-60001	Rx Cool Vent Hdr Isol	YES	NO
SFV-60002	Rx Cool Vent Hdr Iso	YES	NO
SFV-60003	Rx Cool Dr Hdr Iso	YES	NO
SFV-60004	Rx Cool Dr Hdr Iso	YES	NO

SYSTEM: MISCELLANEOUS LIQUID RADWASTE

SFV-66308	Rx Bldg Sump Pump Disc	YES	NO
SFV-66309	Rx Bldg Sump Pump Disc	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: REACTOR COOLANT CHEMICAL ADDITION AND SAMPLING

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
BWS-033	Boric Acid Pump 705A Disc	NO	NO
BWS-034	Boric Acid Pump 705B Disc	NO	NO
BWS-035	Boric Acid Pump 705A Isolation	NO	NO
BWS-036	Boric Acid Pump 705B Isolation	NO	NO
BWS-040	Boric Acid Disch to HPI Pumps	NO	NO
BWS-041	Boric Acid Disch to Decay HT Rem Sys	NO	NO
BWS-054	Boric Acid Disch to P-23BB	NO	NO
RSS-500	Sample to Makeup	YES	NO
PSV-70505	Boric Acid Pump Disch Relief	NO	NO
SFV-70001	Press Liq Sample-Inbd	YES	NO
SFV-70002	Press Sample-Otbd	YES	NO
SFV-70003	Press Vapor Sample-Inbd	YES	NO
SFV-72501	PRT Gas Sample-Inbd	YES	NO
SFV-72502	PRT Gas Sample-Otbd	YES	NO

SYSTEM: TURBINE PLANT SAMPLING

HV-20587	Stm Gen Sample	NO	NO
HV-20588	Stm Gen Sample	NO	NO
HV-20593	Stm Gen Sample	NO	NO
HV-20594	Stm Gen Sample	NO	NO

SYSTEM: BRUCE GM-DIESEL OIL

DFO-017	Diesel FO Pump 888A Disch	YES	NO
DFO-018	Diesel FO Pump 888B Disch	YES	NO
DFO-019	Diesel FO Pump 888C Disch	YES	NO
DFO-020	Diesel FO Pump 888D Disch	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: EMERGENCY DIESEL GENERATOR

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
EGS-010	Air Comp C-889A Disch	YES	NO
EGS-011	Air Comp C-891A Disch	YES	NO
EGS-021	Air Comp C-889B Disch	YES	NO
EGS-022	Air Comp C-891B Disch	YES	NO
FY-89025	Emer Diesel G886A Air Start	YES	NO
FY-89026	Emer Diesel G886B Air Start	YES	NO
FY-89027	Emer Diesel G886A Air Start	YES	NO
FY-89028	Emer Diesel G886B Air Start	YES	NO
FV-89029	Air Supply to Y-890-1C&1D	YES	NO
FV-89030	Air Supply to Y-890-1E&1F	YES	NO
FV-89031	Air Supply to Y-890-1A&1B	YES	NO
FV-89032	Air Supply to Y-890-1G&1H	YES	NO

SYSTEM: TDI DIESEL GENERATOR - TRAIN "A"

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
EGS-843	EDG G-100A Air Start	YES	NO
EGS-845	EDG G-100A Air Start	YES	NO
HV-10051A	EDG G-100A Air Start	YES	NO
HV-10051B	EDG G-100A Air Start	YES	NO
EGS-565	Receiver V-101 C Inlet	YES	NO
EGS-567	Receiver V-101 A Inlet	YES	NO

SYSTEM: TDI DIESEL GENERATOR - TRAIN "B"

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
EGS-844	EDG G-100B Air Start	YES	NO
EGS-846	EDG G-100B Air Start	YES	NO
HV-10050A	EDG G-100B Air Start	YES	NO
HV-10050B	EDG G-100B Air Start	YES	NO
EGS-564	Receiver V-101D Inlet	YES	NO
EGS-566	Receiver V-101B Inlet	YES	NO

TABLE 2 (Continued)

ASME SECTION XI/10 CFR 50.55a IN-SERVICE TESTING PROGRAM

SYSTEM: PLANT AIR

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
SAS-052	Service Air Supply	YES	NO
SAS-054	Service Air Supply	YES	NO

SYSTEM: AUXILIARY GAS

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
NGS-011	N2 Supply to PRT	YES	NO
NGS-017	N2 Sup to Stm Gen	YES	NO
NGS-018	N2 Supply to Stm Gen	YES	NO
SFV-92520	N2 Supp to PRT	YES	NO

SYSTEM: MISCELLANEOUS WATER

VALVE	FUNCTION	TESTED UNDER CURRENT TECH. SPECS.	TESTED UNDER PERMANENTLY DEFUELED TECH. SPECS.
DMW-024	Misc Water to RB	YES	NO
DMW-025	Misc Water to RB	YES	NO

REFERENCES

1. Proposed Amendment No. 182, Revision 0, dated December 28, 1989, letter AGM/NUC 89-269, the Defueled Technical Specifications. This submittal contains the ODCM, REMP Manual, and PCP to support the implementation of Generic Letter 89-01.

2. Generic Letter 89-01, IMPLEMENTATION OF PROGRAMMATIC CONTROLS FOR RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS IN THE ADMINISTRATIVE CONTROLS SECTION OF THE TECHNICAL SPECIFICATIONS AND THE RELOCATION OF PROCEDURAL DETAILS OF RETS TO THE OFFSITE DOSE CALCULATION MANUAL OR THE PROCESS CONTROL PROGRAM

3. Generic Letter 89-14, LINE-ITEM IMPROVEMENTS IN TECHNICAL SPECIFICATIONS - REMOVAL OF THE 3.25 LIMIT ON EXTENDING SURVEILLANCE INTERVALS

4. Generic Letter 88-12, REMOVAL OF FIRE PROTECTION REQUIREMENTS FROM TECHNICAL SPECIFICATIONS

5. Proposed Amendment No. 102 dated December 12, 1984, removal of Appendix B from Operating License No. DPR- 54.

6. Proposed Amendment No. 102, Revision 1 dated July 27, 1988, removal of Appendix B from Operating License No. DPR-54.

7. Proposed Amendment No. 102, Revision 1, Resubmittal dated July 5, 1989, removal of Appendix B from Operating License No. DPR-54.

8. SMUD calculation No. Z-RCS-N0048, Decay Heat Power After 6/7/89 - Cycle 7

9. SMUD calculation No. Z-SFC-N0046, Spent Fuel Heat Generation Following June 7, 1989 Shutdown Of Rancho Seco

10. SMUD Calculation No. Z-SFC-M2533, Time To Boil And Time To Boil Down The Fuel Pool

11. SMUD Calculation No. Z-SFC-N0049, Maximum Predicted Whole Body And Child Thyroid Dose Rates At The Site Boundary From Postulated Accidents During Plant Shutdown

12. SMUD Calculation Z-SFC-0029, Spent Fuel Pool - Surface Dose Rate

13. SMUD to NRC letter AGM/NUC 89-261 dated December 4, 1989, Technical Specifications Applicable In The Defueled Condition

14. SMUD to NRC letter dated April 17, 1989, 10CFR50.63, LOSS OF ALL ALTERNATING CURRENT POWER

REFERENCES (Continued)

15. Regulatory Guide 1.155, STATION BLACKOUT, June 1988

16. Rancho Seco Nuclear Generating Station, Unit No. 1, Updated Safety Analysis Report, Amendment 7

17. NUREG-0103, Rev.4, Standard Technical Specification

18. Proposed Amendment No. 180 dated December 28, 1989, removal of the Fire Protection Requirements from the Technical Specifications per Generic Letter 88-12

19. Proposed Amendment No. 180, supplement 1 dated March 16, 1990, response to NRC comments

20. Amendment No. 115 to Facility Operating License No. DPR-54, Relocation of Fire Protection Program requirements pursuant to Generic Letter 88-12, dated September 10, 1990 (TAC No. 75565)

21. NRC Confirmatory Order dated May 2, 1990 (TAC No. 74592). The NRC placed a License Condition in Facility Operating License DPR-54 which prohibits the District from moving fuel into the reactor building without prior NRC approval.

22. SMUD to NRC letter AGM/NUC 90-114 dated April 26, 1990, Possession-Only License Amendment Request, Proposed Amendment No. 184

23. NRC to SMUD letter dated May 16, 1990, Exemption Related to 10 CFR 55 - Requirements for a Simulation Facility and Simulator Training (TAC No. 75520)

24. NRC to SMUD letter dated May 14, 1990, Waiver of Compliance from Technical Specification 3.2.2, Low Temperature Overpressure Protection

25. SMUD to NRC letter AGM/NUC 90-109 dated May 25, 1990, Refinement To The List Of Technical Specifications Applicable In The Defueled Condition

26. SMUD to NRC letter AGM/NUC 90-212 dated August 7, 1990, Correction To List Of Technical Specifications Applicable In The Defueled Condition

27. NRC to SMUD letter dated February 5, 1990, Technical Specifications Applicable In The Defueled Condition (TAC No. 75395)

28. SMUD to NRC letter AGM/NUC 90-095 dated March 21, 1990, Technical Specifications Applicable In The Defueled Condition - Response To NRC Comments

REFERENCES (Continued)

29. NUREG-0472, Revision 5, Standard Radiological Effluent Technical Specifications

30. NUREG-0103, Revision 4, B&W Standard Technical Specifications, Fall 1980

31. NRC Information Notice No. 90-08 dated February 1, 1990, Kr-85 Hazards from Decayed Fuel

32. NRC Generic Letter 87-09 dated June 4, 1987, Sections 3.0 and 4.0 of the Standard Technical Specifications on the Applicability of Limiting Conditions for Operation and Surveillance Requirements

33. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, July 1980

34. Proposed Amendment No. 182, Revision 1, dated April 30, 1990, letter AGM/NUC 90-130, the Long Term Defueled ~ chnical Specifications. A revised ODCM is provided to support the implementation of Generic Letter 89-01.

ATTACHMENT III

CERTIFIED FUEL HANDLER TRAINING PROGRAMS

CERTIFIED FUEL HANDLER
TRAINING PROGRAMS

Sacramento Municipal Utility District

Rancho Seco Nuclear Generating Station

April 30, 1990

CERTIFIED FUEL HANDLER TRAINING PROGRAMS

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CERTIFIED FUEL HANDLER TRAINING PROGRAMS

1.0 PURPOSE

The purpose of the Certified Fuel Handler Training Programs is to establish the requirements for the initial and proficiency training, testing, certification and recertification of Certified Fuel Handlers.

2.0 SCOPE

The Rancho Seco Nuclear Generating Station is shutdown, defueled and currently storing the nuclear fuel in the spent fuel pool. Implementation of these training programs, consistent with the Long Term Defueled Technical Specifications, ensures a staff trained and certified to safely maintain the spent fuel and perform any fuel movement that may be required.

3.0 REFERENCES

3.1 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, Subpart I - Training and Certification of Personnel

3.2 Rancho Seco Technical Specifications Proposed Amendment No. 182, Rev. 1 - April, 1990

3.3 ANSI/ANS 3.1-1981, American National Standard for Selection, Qualification and Training of Personnel for Nuclear Power Plants

3.4 ANSI N18.1-1971, American National Standard for Selection and Training of Nuclear Power Plant Personnel

4.0 DEFINITIONS

The following terms are defined for uniform interpretation of this document.

4.1 Certified Fuel Handler (CFH) - An individual, certified by the AGM, Nuclear or designee, who manipulates the controls of this nuclear storage facility. A CFH is deemed to manipulate a control if he/she directs another to manipulate a control.

4.2 CFH Initial Training Program - Establishes the prerequisites, structure, content and evaluations required to be successfully completed by candidates for the position of CFH. Successful completion of the program results in initial certification of the individual as a CFH at Rancho Seco.

4.3 CFH Licensed Incumbent (CFHLI) Training Program - Establishes the prerequisites, structure, content and evaluations required to be successfully completed by candidates for the position of CFH. Successful completion of the program results in the certification of a currently NRC licensed operator at Rancho Seco to a CFH at Rancho Seco.

4.4 CFH Proficiency Training Program - Establishes the prerequisites, structure, content and evaluations required to be successfully completed by CFH incumbents. Successful completion of the program results in the recertification of an individual as a CFH at Rancho Seco.

4.5 Controls - Apparatus and mechanisms, the manipulation of which directly affects fuel movement,or the continued safe storage, cooling or shielding of the fuel.

4.6 Examination - A test used to measure knowledge, skills, or ability in one or more subjects or areas. An examination may be a written test or an operating test.

4.7 Nuclear Storage Facility - A complex designed and constructed for the safe storage of spent fuel and other radioactive materials associated with spent fuel storage. The nuclear storage facility at Rancho Seco is comprised of the spent fuel pool and auxiliary systems necessary to safely store the spent fuel.

4.8 Nuclear Storage Facility Experience - For the purpose of the CFH Initial Training Program, experience acquired in any of the following areas shall be evaluated for applicability to qualify as nuclear storage facility experience:

4.8.1 Experience acquired at a nuclear storage facility which involved maintaining spent fuel in a safe storage condition.

4.8.2 Experience acquired in the pre-operational and start-up reactor testing activities, or operation of nuclear power plants including refueling.

4.8.3 Experience in design, construction, and operational training may be considered applicable nuclear storage facility experience.

4.8.4 Experience acquired at military, non-stationary, propulsion, or production nuclear plants may qualify as equivalent to nuclear storage facility experience.

4.9 Shall, Should, May

Shall is used to denote a requirement or commitment.

Should is used to denote a recommendation.

May indicates permission, neither a requirement nor a recommendation.

4.10 Simulate - The training/evaluation method whereby an individual performs a walk-through of actions to be taken, while using the control panel involved and discussing each step without actually operating any controls.

5.0 PROGRAM REQUIREMENTS

5.1 Schedule

5.1.1 The CFH Initial Training Program shall be scheduled and conducted in order to ensure that a sufficient number of CFHs are available to comply with the approved Technical Specifications.

5.1.2 The CFH License Incumbent Training Program shall be scheduled and conducted in order to transition the currently NRC licensed operators at Rancho Seco to CFHs within one year following approval by the NRC of the Rancho Seco Certified Fuel Handler Licensed Incumbent Training Program.

5.1.2.1 If the incumbent licensed operators have not completed this program within one year following NRC approval of the CFH License Incumbent Training Program, then in order to be certified, the individuals shall complete the CFH Initial Training Program.

5.1.3 The CFH Proficiency Training Program shall be conducted for a continuous period not to exceed twenty-four months in duration. Satisfactory completion of the CFH Proficiency Training Program shall be based on successful completion of both the biennial written and annual operating examinations.

5.2 Certification

5.2.1 Certification of fuel handlers shall be granted only by the AGM, Nuclear or designee.

5.2.2 The AGM, Nuclear or designee may grant a certification if one of the following conditions are met:

5.2.2.1 The candidate has successfully completed the CFH Initial Training Program.

5.2.2.2 The candidate has successfully completed the CFH Licensed Incumbent Training Program.

5.2.2.3 The candidate holds a NRC Senior Operator License at Rancho Seco.

5.2.2.4 The candidate has completed the Shift Technical Advisor program and obtained a NRC Operator License at Rancho Seco.

5.2.2.5 The candidate holds a Rancho Seco Senior Operator Certification.

5.2.3 A certification shall typically remain valid for a period of twenty-four months, not to exceed twenty-seven months, from the date of issuance and is contingent on the individual being enrolled and meeting the requirements of the CFH Proficiency Training Program. Within these constraints, a CFH shall successfully complete the CFH Proficiency Training Program within twenty-seven months of initial certification.

5.2.4 The AGM, Nuclear or designee may grant recertification to those participants who have successfully completed the CFH Proficiency Training Program.

5.3 Conditions of Certification

5.3.1 All CFHs shall participate in all of the CFH Proficiency Training Program lecture series. However, up to 30% of the lecture series may be missed because of sickness, emergency shift change or other justifiable cause. After returning to a normal work schedule, any training missed shall require, within three months, satisfactory completion of one or more of the following: documented self-study, one-on-one instruction, or reading assignments; in addition, any examinations missed shall require, within three months, satisfactory completion. If the training is not satisfactorily completed, within the established time limit, the individual shall be suspended from CFH duties. The requirements established in Section 5.3.5 shall be met prior to the resumption of CFH duties.

5.3.2 The requirement for maintaining an active CFH certification shall be to actively perform the functions of a CFH for a minimum of 4 hours per calendar quarter.

5.3.2.1 Failure by a CFH to meet the requirements of Section 5.3.2 shall result in the inactivation of that CFH's certification.

5.3.3 Any CFH who has not been performing the function of a CFH as stated in Section 5.3.2 shall meet the following requirements prior to the resumption of CFH duties:

5.3.3.1 The qualification and status of the certification are current and valid.

5.3.3.2 The inactive CFH has completed a minimum of 12 hours of shift functions under the direction of a CFH and in the position to which the individual shall be assigned. The 12 hours shall have included a review of the spent fuel cooling system, other systems required to support maintaining safe storage of the fuel in the spent fuel pool and all required shift turnover procedures.

5.3.4 A CFH shall be suspended from the CFH duties and subject to the requirements of Section 5.3.5 when:

5.3.4.1 the grade on the biennial written examination is less than 80% overall.

5.3.4.2 a failure grade is received on the annual operating examination.

<u>OR</u>

5.3.4.3 an evaluation made by Operations management indicates the need for remedial training.

5.3.5 Resumption of CFH Duties Following Suspension

5.3.5.1 Prior to resuming the duties of a CFH, the AGM, Nuclear shall establish a review board consisting of an Operations superv or and a training representative.

5.3.5.2 The board shall evaluate and review the individual's past performance and deficiencies.

5.3.5.3 The board may recommend remedial training, a waiver of the suspension or permanent removal from CFH duties.

5.3.5.4 Remedial training and re-examination should be successfully completed within 30 days of the board's recommendation. If remedial training and re-examination is not completed within 60 days, the Board shall reconvene and evaluate the circumstances. Approval by the AGM, Nuclear, is required if the Board recommends further training and re-examination.

5.3.5.5 If the individual fails the second round of re-examination, their certification shall be revoked. The individual may be recertified only after successful completion of the CFH Initial Training Program.

5.4 Evaluations

5.4.1 The CFH Initial, Proficiency, and Licensed Incumbent Training Programs shall include an evaluation process to obtain maximum benefit from the programs and to ensure CFH competence.

5.4.2 Successful completion of any written examination shall be measured by the achievement of 80% or better on that examination.

5.4.3 The operating examinations shall be graded on a pass/fail basis.

5.5 Medical Requirements

5.5.1 Candidates for the CFH Initial Training Program shall have a medical examination by a physician.

5.5.2 All CFHs shall have a medical examination by a physician every two years.

5.5.3 The physician shall determine that the medical condition and general health of the CFH candidate or incumbent CFH should not adversely affect the performance of assigned CFH duties or cause operational errors endangering public health and safety.

6.0 CERTIFIED FUEL HANDLER INITIAL TRAINING PROGRAM

6.1 Prerequisites

6.1.1 Candidates for the CFH Initial Training Program shall have a high school diploma or equivalent.

6.1.2 Candidates for the CFH Initial Training Program shall have a minimum of one year of nuclear storage facility experience.

6.2 Program Structure

6.2.1 The CFH Initial Program is established to provide the training, testing and certification for replacement CFHs.

6.2.2 The CFH Initial Training Program is divided into three training phases. They are the Classroom training phase, the On-the-Job training phase and the Self-Study training phase.

6.2.3 The Classroom training phase should comprise approximately thirty percent of the CFH Initial Training Program and shall include a series of training courses in spent fuel storage fundamentals, systems, procedures and supervisory skills.

6.2.3.1 The Classroom training phase is designed to meet the course objectives by utilizing preplanned lectures, seminars, and supervised discussions.

6.2.3.2 The Classroom training phase may include the use of films, video tapes, programmed instruction and other effective training aids to supplement the lectures; however, their use shall be limited to less than fifty percent of the total Classroom training phase.

6.2.4 The On-the-Job training phase should comprise approximately fifty percent of the CFH Initial Training Program and shall include the application of the knowledge gained during the Classroom training phase of the program.

6.2.4.1 The On-the-Job training phase shall consist of a checkout book to document tasks performed or simulated.

6.2.5 The Self-Study training phase should comprise approximately twenty percent of the CFH Initial Training Program and should include additional information which supplements the Classroom training phase lectures.

6.2.5.1 An instructor shall be available to monitor progress and answer questions during the Self-Study training phase.

6.2.6 A candidate should complete the CFH Initial Training Program within twelve months after entering the program.

6.3 Program Content

6.3.1 The Classroom training phase shall cover the following:

a) Nuclear Theory associated with spent fuel storage

b) Heat Transfer and Fluid Flow associated with heat removal from the spent fuel

c) Radiation Control and Safety

d) Design characteristics and the operations of the spent fuel handling and storage equipment

e) Electrical systems associated with the nuclear storage facility

f) Plant auxiliary and support systems associated with the nuclear storage facility

g) Normal, casualty and emergency procedures

h) Administrative controls applicable to the nuclear storage facility

i) Applicable industry events and lessons learned

j) Supervisory training for CFH

k) Fundamentals in science (for example, chemistry, electrical theory and material science, as deemed appropriate)

6.3.2 The Self-Study training phase shall cover the following:

a) Licensing basis documents such as the Technical Specifications and the Updated Safety Analysis Report (USAR)

b) A review of procedures required for the safe operation of the nuclear storage facility

6.3.3 The On-the-Job training phase shall cover a review of the knowledge gained in the Classroom training phase by simulation and actual performance of the CFH's duties.

6.3.3.1 The On-the-Job training phase shall cover tasks such as spent fuel bridge checkout, spent fuel pool water addition, etc.

6.4 Evaluations

6.4.1 A written examination shall be given to the candidates upon completion of each CFH topical course.

6.4.2 Upon completion of all three phases of the CFH Initial Training Program, a comprehensive written examination shall be administered to determine the individual's knowledge to safely and competently maintain the fuel in the spent fuel pool.

6.4.3 Upon completion of all three training phases of the CFH Initial Training Program, an operating examination shall be administered to demonstrate the individual's ability to perform the duties of a CFH.

6.4.4 Satisfactory completion of the CFH Initial Training Program and subsequent certification shall be based on the successful completion of both the comprehensive written and operating examinations.

7.0 CERTIFIED FUEL HANDLER PROFICIENCY TRAINING PROGRAM

7.1 Prerequisites

7.1.1 Participants in the CFH Proficiency Training Program are CFHs as described in Section 5.2.

7.2 Program Structure

7.2.1 The CFH Proficiency Training Program is established to maintain the proficiency of CFHs.

7.2.2 The CFH Proficiency Training Program is divided into three training phases. They are the Classroom training phase, the On-the Job training phase and the Self-Study training phase.

7.3 <u>Program Content</u>

7.3.1 The CFH Proficiency Training Program is designed to review the contents of the initial training program with the exception of introductory fundamental training.

7.3.2 The CFH Proficiency Training Program is designed with enough flexibility such that training may be offered on the activities planned or in progress and on identified CFH weaknesses.

7.3.3 The following topics shall be reviewed during the two year proficiency training period:

a) Nuclear Theory associated with spent fuel storage

b) Heat Transfer and Fluid Flow associated with heat removal from the spent fuel

c) Radiation Control and Safety

d) Operation of the spent fuel handling and storage equipment

e) Plant auxiliary and support systems associated with the nuclear storage facility

f) Normal, casualty, and emergency procedures

g) Administrative controls applicable to the nuclear storage facility

h) Applicable industry events and lessons learned

i) Licensing basis documents such as the Technical Specifications and the Updated Safety Analysis Report (USAR)

7.4 Evaluations

7.4.1 Written examinations shall be given to the CFHs at the completion of a block of training. Blocks of training occur throughout the two year proficiency period based on the CFHs' shift rotation.

7.4.2 A comprehensive written examination shall be administered on a biennial basis to determine the individual's knowledge to continue to safely and competently operate the nuclear storage facility.

7.4.2.1 The examination shall be used as a diagnostic instrument to determine the CFH's knowledge of subjects covered in the proficiency program and as a method to determine areas in which retraining is needed to upgrade the CFH's knowledge.

7.4.2.2 Individuals involved with the preparation, review and grading of the biennial written examination (up to a maximum of 3) shall be exempted from that examination.

7.4.2.3 No individual shall be exempted from two consecutive written biennial examinations.

7.4.3 An annual operating examination shall be administered to demonstrate the individual's ability to perform the duties of a CFH.

7.4.4 Satisfactory completion of the CFH Proficiency Training Program and subsequent recertification shall be based on the successful completion of both the comprehensive written and operating examinations.

7.4.5 Should the performance of the CFH be deemed unsatisfactory, the CFH shall be suspended from the CFH duties and shall meet the requirements of Section 5.3.5 prior to the resumption of CFH duties.

8.0 CERTIFIED FUEL HANDLER LICENSED INCUMBENT TRAINING PROGRAM

8.1 Prerequisites

8.1.1 The candidate for certification is currently NRC licensed as an operator at Rancho Seco.

8.2 Program Structure

8.2.1 The CFH Licensed Incumbent (CFHLI) Training Program is established to provide training to the licensed operators who have not previously obtained supervisory training.

8.2.2 The CFHLI Training Program is divided into two training phases. They are the Classroom training phase and the On-the-Job training phase.

8.2.3 The Classroom and On-the-Job training phases should comprise approximately twenty-five and seventy-five percent respectively of the CFHLI Training Program.

8.2.4 A candidate should complete the CFHLI Training Program within two months after entering the program.

8.3 Program Content

8.3.1 The CFHLI Training Program shall provide supervisory training for all operators who have not previously obtained supervisory training.

8.3.2 The supervisory training shall provide training on the following skills:

a) Leadership

b) Interpersonal communication

c) Command responsibilities and limits

d) Motivation of personnel

e) Problem analysis

f) Decision analysis

g) Supervisor administrative requirements

8.3.3 All licensed operators at Rancho Seco are exempt from the CFH Initial Training Program since this material is contained in the Licensed Operator Training Programs and the CFHLI Training Program.

8.4 Evaluations

8.4.1 Upon successful completion of the CFHLI Training Program, an operating examination shall be administered to demonstrate the individual's ability to perform the duties of a CFH.

8.4.2 Satisfactory completion of the CFHLI Training Program and subsequent certification shall be based on the successful completion of the operating examination.

9.0 INSTRUCTOR QUALIFICATIONS

9.1 Instructors shall be certified by the AGM, Nuclear.

9.2 The CFH instructor's certification shall be based on technical knowledge of the Rancho Seco nuclear storage facility and instructional qualifications.

9.2.1 Instructors qualified to teach the Licensed Operator Requalification Training Program are qualified to instruct in the CFH training programs.

9.3 CFH instructors shall be enrolled in the CFH Proficiency Training Program to assure they are cognizant of current operating history problems and changes to procedures and administrative limitations.

ATTACHMENT IV

OFFSITE DOSE CALCULATION MANUAL

REVISION SUMMARY:

1. For general Revision Summary, see Attachment 1 of 50.59 Review.

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1.0 PURPOSE

The Offsite Dose Calculation Manual (ODCM) contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents. Also, the ODCM contains the methodology for determining effluent monitoring instrumentation alarm/trip setpoints. Methods are described for assessing compliance with the Technical Requirements in the ODCM as they apply to 10 CFR Part 20.106, 10 CFR Part 50, Appendix I, and 40 CFR 190.10 for liquid and gaseous effluents. Additionally, the ODCM contains the Technical Requirements which provide the Specifications, Applicabilities, Actions, and Surveillance Requirements.

2.0 SCOPE

This procedure functions as a manual that provides the basis for development of detailed implementing procedures that address dose calculations for liquid/gaseous releases and monitor setpoints.

3.0 REFERENCES/COMMITMENT DOCUMENTS

3.1 Commitment Documents

3.1.1 Code of Federal Regulations, Title 10, Chapter 1, Parts 20, 50.36a and Part 50, Appendix I

3.1.2 Rancho Seco Technical Specifications

3.1.3 CCTS T890828001, EPA 40 CFR Parts 302, 355 Reporting Requirements

3.1.4 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Plant Operations

3.2 Reference Documents

3.2.1 USNRC Regulatory Guide 1.109, Rev. 1, 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, October 197[illegible]

3.2.2 W.C. Burke, et.al., Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, NUREG-0133, USNRC:NRR, October [illegible]

3.2.3 ORNL, User's Manual for LADTAP II, NUREG/CR-1276, May 1980

3.2.4 D.L. Strange, et.al., LADTAP-II, Technical Reference and User Guide, NUREG/CR-4013, Pacific Northwest Laboratory, April 1986

3.2.5 Eckerman, K.F., et.al., User's Guide to GASPAR Code, NUREG-0597, USNRC:NRR, June 1980, in RSIC CCC-463

3.2.5 USNRC Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

3.2.7 USNRC 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

3.2.8 USNRC Regulatory Guide 4.1, Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants

3.2.9 REIMS Software Life Cycle Documents (Software Requirement Specification, Design Document, Acceptance Test Plan)

3.2.10 USNRC & Pacific Northwest Laboratory, TDMC Computer Code/Data Collections, XOQDOQ-82, Radiological Assessment Code System Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations

3.2.11 1988 Rancho Seco Land Use Census

3.2.12 RSNGS U : R Chapters 11.1-11.5

3.2.13 RSNGS P&ID Drawing M-563, M-551, M-552, M-503

3.2.14 Pacific Northwest Laboratory, XOQDOQ: Computer Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations

3.2.15 Congel, F.J., Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40 CFR Part 190), NUREG-0543, USNRC:NRR, February 1980

3.2 16 USNRC Generic Letter, 89-01, Dated January 31, 1989.

3.2.17 Rancho Seco REMP Manual

4.0 DEFINITIONS

4.1 Member(s) of the Public

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

4.2 Batch Release

A Liquid Batch Release for 10 CFR 50 Appendix I considerations is a transfer of a discrete volume of radioactive liquid from a RHUT to a retention basin. A Liquid Batch Release for 10 CFR 20 considerations is a transfer of a discrete volume of radioactive liquid from a retention basin to the Waste Water discharge canal (the Environmental Release Point).

A Gaseous Batch Release is the discharge of gaseous radioactive wastes of discrete volume. Batch releases for the gaseous pathway include initial Reactor Building purges.

An initial RB purge includes all the radioactive gas released from the first opening of the containment until the RB atmosphere concentration of activity reaches a new equilibrium (normally seven volume changes) due to leakage of Reactor Coolant during cold shutdown. Further release from this pathway is called a continuing Reactor Building purge and considered a continuous release.

4.3 Continuous Release

A continuous radioactive gaseous release is the discharge of gaseous wastes of a nondiscrete volume from a system that may have an input flow during the release. These include the Auxiliary Building Stack (ABS), Auxiliary Building Grade Level Vent (ABGLV), and continuing Reactor Building purges (see Definition 4.2 above).

Continuous radioactive liquid releases are not planned to be made from Rancho Seco Nuclear Generating Station (RSNGS).

4.4 Default Radionuclide Mix

A historical mixture of radionuclides that may be used to determine monitor setpoints.

4.5 Dilution Flow

The volume or volume rate of fluid (liquid or gas) which is added to a radiolcgical release stream for the purpose of decreasing the instantaneous concentration of the stream.

4.6 Maximum Exposed (Hypothetical) Individual

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

Maximum dose factor parameters will be determined using site specific data from the Land Use Census. If information needed to determine a parameter is not available, RG 1.109 parameters will be used. All dose factor parameters used are listed in Attachment 3.

4.7 RSNGS

Rancho Seco Nuclear Generating Station.

4.8 Unrestricted Area

Any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area within the site boundary used for residential quarters or industrial, commercial, institutional and recreational facilities. For environmental protection/effluent control purposes, the Rancho Seco Unrestricted Area is the area outside the Site Boundary as shown in Attachment 5.

4.9 Site Boundary

The Site Boundary for Gaseous Effluents is defined by the drawing in Attachment 5. The Boundary for Liquid Effluents is defined by the drawing in Attachment 6. These boundaries are for 10 CFR 20 and 10 CFR 50 compliance.

4.10 Nuisance Pathways

(1) Secondary system gaseous pathways where the calculated dose totals contribute less than 5% of the annual limits and do not need to be tracked for dose calculational purposes unless secondary activity reaches a predetermined Action Level.

(2) Sources of trace levels of radioactivity in liquid effluents where the calculated dose totals contribute less than 1% of the annual limits and do not need to be tracked for dose calculational purposes. Trace levels are defined to be less than 1E-8 µCi/ml for the nuclides typically released from RSNGS. Examples include the oily water separator, plant effluent inlet, and storm drains.

4.11 Unplanned Release

The unexpected release of radioactive materials to unrestricted areas in gaseous and liquid effluent. All unplanned releases shall be discussed in the Semi-annual Radiological Effluent Release Report to the NRC.

4.12 Miscellaneous Release

Release pathways which are considered planned but are not defined explicitly with monitoring requirements in Technical Specifications. These pathways contribute a relatively small percentage (<5%) to the annual dose limits but shall be tracked for effluent activity accounting and dose calculation purposes. Miscellaneous releases shall not be reported in the SAER as abnormal or unplanned releases. Examples include the IOSF and the Respirator Cleaning Facility.

4.13 Safety Factor (SF)

A number greater than unity used in calculations to introduce greater conservatism (larger margin of safety) to offset various uncertainties in instrumentation and methods. Safety factors are set by Chemistry Supervision based on either analysis or professional judgement. Unless otherwise specified, the default value is two (2).

4.14 Liquid Effluent Radwaste Treatment System

The Liquid Effluent Radwaste Treatment System refers to the sluiceable demineralizer skid (see Drawing M-563 Sheet 3) designed and installed to reduce the quantity of radioactive materials in liquid effluents.

4.15 Gaseous Radwaste Treatment System (GRTS)

The term Gaseous Radwaste Treatment Systems refers to the Ventilation Exhaust Treatment System.

4.16 Ventilation Exhaust Treatment Systems

The Ventilation Exhaust Treatment Systems, which are the Reactor Building Purge Exhaust Filtering System and Auxiliary and Spent Fuel Building Filter Systems, are systems designed and installed to reduce radioactive material in exhaust gases through HEPA filters for the purpose of removing particulates from the gaseous exhaust stream prior to the release to the environment (such a system is not considered to have any effect on noble gas effluents). Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be Ventilation Exhaust Treatment Systems components.

4.17 Instrument Surveillance

(1) Trip Test

A trip test is a test of logic elements in a protection channel to verify their associated trip action.

(2) Channel Test

A channel test is the injection of an internal or external test signal into the channel to verify its proper response, including alarm and/or trip initiating action, where applicable.

(3) Instrument Channel Check

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable.

(4) Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm, or trip and shall be deemed to include the channel test.

4.18 Surveillance Intervals

The Surveillance Interval may be extended to a maximum of +25% to accommodate operations scheduling. The frequency notation (which follows the name of the Surveillance Interval in parenthesis) specified for the performance of Surveillance Requirements shall correspond to the Surveillance Intervals defined below.

(1) Shift (S): A time period covering at least once per twelve (12) hours.

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

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

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

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

(6) Semi-Annually (SA): A time period spaced to occur at least once per six (6) months.

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

(8) Refueling Interval (R): A time period spaced to occur at least once per eighteen (18) months.

(9) Each Release (P): This surveillance will be completed prior to each release.

4.20 Reactor Operating Conditions

(1) Hot Shutdown

The reactor is in the hot shutdown condition when $k_{eff} \leq 0.99$, and T_{avg} is at or greater than 525°F.

(2) Power Operation

The reactor is in a power operation condition when the indicated neutron power is above 2 percent of rated power as indicated on the power range channels.

5.0 **RESPONSIBILITIES**

5.1 It is the responsibility of the Chemistry Manager for the following:

(1) ODCM Revisions and Reporting the Revisions in the SRERR (Semi-annual Radioactive Effluent Release Report)
(2) SRERR Preparation and Submittal
(3) REIMS Database
(4) Meteorological Database
(5) LADTAP, GASPAR, and XOQDOQ Computer Program Verifications and Changes

5.2 The PRC and MSRC are responsible for reviewing and approving all changes to the ODCM per Technical Specifications.

6.0 **PROCEDURE**

6.1 General Considerations

6.1.1 Liquid Effluent Pathways

Attachment 1 provides a simplified diagram of the radioactive liquid effluent produced by RSNGS. The liquid effluent discharge of RSNGS forms the headwaters of Clay Creek.

Dilution of the liquid effluent occurs offsite at the confluence of Clay and Hadselville Creeks, and of Hadselville and Laguna Creeks, and at the confluence of Laguna Creek and the Cosumnes River.

Planned radioactive liquid releases from secondary plant systems are directed through the A or B RHUTs to give reasonable assurance of compliance with 10 CFR 50 Appendix I prior to their discharge to the retention basins (North or South). Prior to discharge from the retention basins to the plant effluent (offsite), the discharge rate from the retention basins and the amount of dilution from Folsom South Canal are controlled to ensure compliance with the concentration requirements in 10 CFR 20.

6.1.2 Gaseous Effluent Pathways

Airborne radioactive material in the various rooms and systems at RSNGS is routed and discharged in airborne effluent as illustrated schematically in Attachment 2. The figure shows the functional arrangements of these streams, treatment and controls, radioactivity monitoring points, and effluent release points.

Before the gas in the reactor building (initial purge) is released, noble gas concentrations are determined and pre-release dose and dose rate calculations are done in accordance with the methodology of Steps 6.8 and 6.9. These measured concentrations are used to establish maximum effluent discharge rates and monitor setpoints.

Other potential release pathways than those specified in Attachment 2 have been identified. These release pathways are classified as NUISANCE pathways and include the following:

(1) Auxiliary Boiler Vents/Reliefs
(2) Miscellaneous Secondary System Steam Discharges
(3) Tank Atmospheric Vents
(4) Condenser Hogging Air Ejectors
(5) Routine Secondary Plant Steam Leakage
(6) Auxiliary Feedwater Pump Turbine Exhaust
(7) Service Air/Nitrogen Supply System

Past experience has shown that the above release pathways do not contribute to the dose totals because of the small quantities released and the low concentration of radioactive materials. Therefore, Action Levels may be established for the secondary system concentrations of radioactive material to trigger when the above routine gaseous effluent releases shall be evaluated for offsite dose impact. The Action Levels shall be based on levels that could contribute more than 5% to the most restrictive yearly dose limit. Action Levels shall be maintained through RSNGS procedures.

Unplanned releases shall be evaluated on a case by case basis.

The Respirator Cleaning Trailer and Interim Onsite Storage Facility (IOS) are miscellaneous releases.

6.1.3 Meteorological Data

Atmospheric dispersion (X/Q) and deposition factors (D/Q) used in performing calculations involving airborne effluent are derived via the MIDAS program from meteorological measurements made at RSNGS meteorological tower. The measurements are reduced to atmospheric stability classes and joint frequency tables. Using these tables, relative atmospheric dispersion and deposition factors for a ground-level, building wake discharge are computed with a straight line, 22.5 degree, cardinal sector-width-averaged, Gaussian plume diffusion model.

Several years of meteorological data collected at RSNGS have been reduced to the relative atmospheric dispersion and deposition factors in Attachment 4. These factors should be used to determine monitor setpoints and to assess compliance with Section 6.14 Specifications. Actual meteorological data for the periods of release shall be used to calculate dose in the SRERR.

6.1.4 Boundaries

The boundary for compliance with gaseous effluent Specifications, per Section 6.14 Technical Requirements, is the Site Boundary for Gaseous Effluents, as shown on Attachment 5. This boundary is used for calculations involving the Specifications in Steps 6.14.6, 6.14.7, 6.14.8, and MPC calculations for compliance with 10 CFR 20.106 and 10 CFR 50.72.

The Boundary for Liquid Effluents as shown in Attachment 6 is for calculations involving the Specification in Step 6.14.2 (liquid pathway).

6.1.5 40 CFR 190 Compliance

For the purposes of assessing compliance with 40 CFR 190, the MEMBER OF THE PUBLIC which received the most exposure may be determined using actual food consumption, actual occupancy rates, and dilution offsite from additional converging streams (verses assumptions used for a HYPOTHETICAL MAXIMUM EXPOSED INDIVIDUAL based on Land Use Census data).

6.1.6 Computers vs Manual Calculations

Computer systems such as REIMS should be used for calculations in order to minimize error and hasten the release process. However, in the event computers are not available for calculations, manual pre-release calculations should be done based on the most historically restrictive receptor.

6.2 Liquid Monitor Setpoints

The High alarm setpoint for the Retention Basin Effluent Discharge Monitor (R15017A) is set to ensure the limits of the Specification in Step 6.14.2 are not exceeded. When the high alarm level is reached, the release of the basin is terminated automatically.

A SAFETY FACTOR is included in the setpoint calculations to incorporate a margin of conservatism.

When a batch release is not occurring or the calculated setpoint is so low that it will cause spurious alarms, the monitor setpoint should be set as close to background without causing spurious alarms or as determined by Chemistry Supervision.

The conversion factor and setpoint calculations should be performed based on the same radionuclide mix.

6.2.1 Conversion Factors for R15017A

Provided here is the methodology to determine the conversion factor of counts per minute to microcuries per cubic centimeter for the Retention Basin Radiation Monitor (R15017A). The conversion factor is based on the monitor's efficiency for each nuclide and the abundance of the nuclide. The mix of isotopes used may be based on the historical mix provided in Attachment 6, current mix in the batch release, or as determined by Chemistry Supervision. The mix fraction shall be based on gamma emitting isotopes only.

The following equation shall be used to determine the conversion factor for R15017A:

$$CF = [\ \Sigma_i (f_i)\ (E_i)]^{-1}$$

Where:

CF = µCi/cc per cpm

f_i = Fraction of nuclide i to total activity of historical mix (Attachment 7) or batch mix

E_i = Efficiency for nuclide i (cpm/µCi/cc) Attachment 8

6.2.2 High Alarm Setpoint for R15017A (µCi/ml)

$$\text{HIGH ALARM } (\mu Ci/ml) = \frac{\Sigma C_i}{SF * \Sigma \dfrac{C_i}{MPC_i}} + C_{bkg}$$

Where:

ΣC_i = The summation of all gamma-emitting nuclide concentrations (µCi/ml).

$\Sigma \dfrac{C_i}{MPC_i}$ = The summation of the ratios of all nuclide concentrations to their MPC values. The most limiting MPC values should be selected (soluble or insoluble) for water in unrestricted areas.

SF = A SAFETY FACTOR which may be applied to incorporate a margin of conservatism ($SF \geq 1$).

C_{bkg} = The background reading of the monitor (μCi/ml).

6.3 Maximum Permissible Concentrations

The Maximum Permissible Concentration Fraction is calculated to determine compliance with 10 CFR 20 requirements and the Specification in Step 6.14.2. Radioactive liquid effluent discharges normally originate in the RHUTs and are discharged into a retention basin. Samples are collected and analyzed from each Retention Basin prior to discharge to ensure that compliance with the Specification in Step 6.14.2 can be achieved.

In addition, calculations to determine the minimum dilution water flow rate and maximum retention basin discharge flow rate to ensure compliance are provided in this section. Any combination of minimum dilution flow rate and maximum discharge flow rate which satisfy the Specification is acceptable.

6.3.1 Maximum Permissible Concentration Fraction (MPCF)

Compliance with the Specification in Step 6.14.2 is anticipated when the MPCF is less than or equal to 1.0. The MPCF is calculated as follows:

$$MPCF = \Sigma_i \left[\frac{C_i}{MPC_i} \right] * \frac{F_r}{F_c + F_r}$$

Where:

MPCF = The calculated fraction of Maximum Permissible Concentration in the radioactive liquid effluent discharged into the unrestricted area.

C_i = The total concentration of radionuclide i in the batch, prior to dilution, of liquid effluent in μCi/ml.

MPC_i = The MPC of radionuclide i from Appendix B, Table II, Column 2 of 10 CFR 20 in μCi/ml. The most limiting MPC value should be selected (soluble or insoluble).

F_r = Discharge flow rate; the flow rate of the radioactive liquid batch release from the retention basin to the Waste Water Discharge Canal (Plant Effluent) in gpm.

F_c = The total available dilution water (Plant Effluent) flow rate at the time of discharge of the radioactive liquid effluent in gpm.

6.3.2 The minimum dilution water (Plant Effluent) flow rate (F_{cmin}) is calculated as follows:

$$F_{cmin} = F_r \left[SF \,\Sigma_i \left\{ \frac{C_i}{MPC_i} \right\} - 1 \right]$$

Where:

F_r = A fixed effluent discharge flow (gpm) (as required by specific release restrictions).

SF = A factor which may be applied to incorporate a margin of conservatism ($SF \geq 1$).

NOTE

$SF\ \Sigma_i (C_i/MPC_i)$ must be > = 1.

6.3.3 The maximum effluent discharge flow rate (F_{rmax}) is calculated as follows:

$$F_{rmax} = \frac{F_c}{SF\ \Sigma_i \left\{ \frac{C_i}{MPC_i} \right\} - 1}$$

Where:

F_c = A fixed dilution water flow rate (gpm) (as required by specific release restrictions).

NOTE

$SF\ \Sigma_i (C_i/MPC_i)$ must be > = 1.

6.4 Liquid Dose Calculations

This section provides the methodology to demonstrate compliance with the Specification in Step 6.14.3.

Site specific organ dose factors for liquid effluents have been determined for the MAXIMUM EXPOSED INDIVIDUAL and are listed in Attachment 9. Dose factors (A_{ijap}) were derived using equations and methods in Regulatory Guide 1.109, Rev. 1 and LADTAP. The dose factor parameters used are listed in Attachment 3. As previously stated, site specific parameters should be used based on the Land Use Census in lieu of the values provided in RG 1.109 whenever possible.

The exposure pathways included in the A_{ijap} are those identified by the Land Use Census. The pathways considered for inclusion are:

(1) fresh water fish
(2) fresh water invertebrate
(3) river shoreline deposits
(4) milk from cows that eat fresh or stored forage irrigated with Clay Creek water
(5) meat from cows that eat fresh or stored forage irrigated with Clay Creek water
(6) vegetation

6.4.1 Liquid Effluent Dose Equation

$$D_{aj} = \frac{\Sigma_i \ \Sigma_p \ Q_i \ * \ A_{ijap}}{F}$$

Where:

D_{aj} = Annual calculated dose (50 year dose commitment) to the organ (or total body) j of a maximally exposed individual of age group a (mrem/yr).

Q_i = Activity of isotope i released during the year (Ci/yr).

A_{ijap} = Site specific dose factor for an organ (or total body) j for a person of age group a via pathway p due to isotope i (mrem-ft^3/Ci-sec).

F = Annual average discharge volumetric flow rate (effluent water plus dilution water) in ft^3/sec.

Because the dose rate varies linearly with activity release rate, the dose for a shorter period of time (mrem) may be calculated by substituting the activity released (Ci) during that period for Q_i in the above equation. However, volumetric flow rates should not be averaged over a period less than a calendar quarter. More conservative flow rates are acceptable.

6.5 Liquid Dose Projection

31-day dose projections are calculated to show compliance with the Specification in Step 6.14.4. The 31-day dose projections do not need to be performed when the LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM (LERTS) is operable. The LERTS is considered operable by meeting the Specifications in Steps 6.14.2 and 6.14.3.

If the 31-day dose projection exceeds the limits of the Specification in Step 6.14.4, then either of the following actions should be taken until projections fall within limits:

(1) Require the use of LERTS for every RHUT release when activity can be reduced.

(2) Reduce blowdown from systems known to be significantly contaminated.

The following equations shall be used:

31-Day Projection:

$D_{p31} = 31 * (D_{Yr} / t_{Yr})$

Quarterly Projection:

$D_{pQtr} = 91.3 * (D_{Qtr} / t_{Qtr})$

Yearly Projection:

$D_{pYr} = 365.25 * (D_{Yr} / t_{Yr})$

Where:

D_{p31} = 31-day Projected dose.

D_{Yr} = Cumulative annual dose to date.

t_{Yr} = Number of days into the year.

D_{pQtr} = Quarterly dose projection.

D_{Qtr} = Cumulative quarterly dose to date.

t_{Qtr} = Number of days into the quarter.

D_{pYr} = Annual dose projection.

6.6 Gaseous Monitor Setpoints

The Specification in Step 6.14.5 states that the gaseous effluent monitors shall have their alarm/trip setpoints set to ensure the limits of the Specification in Step 6.14.6 are not exceeded. Data from the sector where the most conservative atmospheric dispersion (maximum X/Q) factor at the Site Area Boundary occurs is used. This value is obtained from Attachment 4. Compliance with the dose rate limits for noble gases in the Specification in Step 6.14.6 is demonstrated by setting each gaseous effluent monitor alarm/trip setpoint so that an alarm/trip will occur at or before the dose rate limit is reached.

A SAFETY FACTOR is included in the setpoint calculations to incorporate a margin of conservatism.

Maximum design flow rates for each release point will be used to calculate setpoints.

6.6.1 Conversion Factors for R15044, R15045, and R15546A

Provided here is the methodology to determine the conversion factor of counts per minute to microcuries per cubic centimeter for the Auxiliary Building Stack Monitor (R15045), Reactor Building Stack Monitor (R15044), and the Auxiliary Building Grade Level Vent Monitor (R15546A). The conversion factor is based on the monitor's efficiency of detection for each nuclide and the mix of the nuclide. The mix of isotopes used may be based on the historical mix provided in Attachment 10, the current mix in the batch or continuous release, or as determined by Chemistry Supervision. The mix used for setpoint calculations and conversion factor calculations should be the same.

The following equation shall be used to determine the conversion factor for R15044, R15045 and R15546A:

$$CF = [\ \Sigma_i (f_i)\ (E_i)]^{-1}$$

Where:

CF = µCi/cc per cpm

f_i = Fraction of nuclide i to total activity or historical mix (Attachment 10) or current mix

E_i = Detector sensitivity to nuclide i, in cpm per µCi/cc, Attachment 11

6.6.2 Gaseous effluent release point and maximum design flow rates used at RSNGS are as follows:

Reactor Building Stack	74,000 CFM
Auxiliary Building Stack	47,740 CFM
Auxiliary Building Grade Level Vent	21,450 CFM
Interim Onsite Storage Building Ventilation*	8,050 CFM
Respirator Cleaning Facility*	1,300 CFM

* The Interim Onsite Storage Building (IOS) and Respirator Cleaning Facility are not Technical Specification release pathways and are not subject to continuous discharges of radioactivity. Because of the infrequency of a radioactive release, assessment will be done on each release according to administrative procedures.

6.6.3 Determination of Partition Factor (P_v)

The Specification in Step 6.14.6 applies to the entire site, not just one vent or monitor. Consequently total release rate must be partitioned among the three major vents (ABS, ABGLV, RBS). For routine operations, the partition factor may be calculated by assuming that the effluent concentration is the same for all pathways and using a ratio of flow rates. The total volume flow rate for all three vents is 143,190 CFM.

Therefore:

$$P_{rbs} = \frac{74,000 \text{ CFM}}{143,190 \text{ CFM}} = 0.52$$

$$P_{abs} = \frac{47,740 \text{ CFM}}{143,190 \text{ CFM}} = 0.33$$

$$P_{abglv} = \frac{21,450 \text{ CFM}}{143,190 \text{ CFM}} = 0.15$$

Chemistry Supervision may elect to use a different set of partition factors based on plant conditions. However, the sum of all the partition factors for the site must be less than or equal to unity (1).

6.6.4 Channel 4 Noble Gas Setpoint for R15044, R15045, and R15546A in µCi/sec

$$M_v = \frac{500 * P_v * \Sigma_i C_i}{SF * (X/Q) * \Sigma_i (C_i * K_i)} + Bkgd$$

Where:

M_v = Monitor setpoint for vent v (i.e., RBS, ABS, or ABGLV) in µCi/sec

500 = Step 6.14.6 Specification limit for total body dose rate in mrem/yr

P_v = Partition factor, dimensionless, which distributes the total site release rate among the three vents

C_i = Concentration of isotope i in gaseous effluent in µCi/cc. The mix of isotopes used may be based on the vent specific historical mix provided in Attachment 10 or based on the current mix in the batch or continuous release.

SF = Safety Factor, dimensionless, ($SF \geq 1$)

X/Q = Most conservative atmospheric dispersion factor for a ground level release to a sector beyond the Site Area Boundary in sec/m^3. Either an annual average or multiple-year average is acceptable.

K_i = Factor converting ground-level concentration of noble gas isotope i to total body dose (mrem-m^3/µCi-yr)

Bkgd = Monitor background reading in µCi/sec

6.6.5 Channel 1 Noble Gas Setpoint for R15044, R15045, and R15546A in µCi/cc

$$MC_v = \frac{MR_v}{472 * F_v} + Bkgd$$

Where:

MC_v = Monitor setpoint for vent v based on concentration in µCi/cc

MR_v = Monitor setpoint for vent v based on release rate in µCi/sec excluding the background term. That is, M_v - Bkgd

F_v = Maximum design volumetric flow rate in CFM as indicated in 5.6.2.

472 = 28317 ml/ft^3 * 1 min/60 sec

Bkgd = Monitor background reading in µCi/cc

NOTE

Channel 1 does not cause any automatic terminations or audible alarms.

6.7 Maximum Permissible Concentrations (MPCs) in Gaseous Effluents

In order to demonstrate compliance with 10 CFR 20.106 which requires that the total MPC fraction not exceed 1 when averaged over an entire year, the calculation is included in the Semi-Annual Effluent Report. In addition, a four hour reporting requirement exists when the total MPC fraction exceeds two when averaged over one hour per 10 CFR 50.72. The following provides guidance on how to perform this calculation.

Maximum Permissible Concentration Fraction (MPC_f) Equation

$$MPC_f = \Sigma \frac{(C_i * F * 4.72E\text{-}4)}{MPC_i} * X/Q * TR$$

Where:

C_i = The concentration of nuclide i in μCi/cc.

F = The flowrate of the effluent release point in ft^3/min.

X/Q = The maximum noble gas X/Q at the Site Area Boundary in sec/m^3.

MPC_i = The most restrictive MPC for nuclide i from 10 CFR 20.106, Appendix B, Table II, Column 1 (μCi/cc).

TR = If the time of release is less than one hour, then this value is the duration of the transient in minutes divided by sixty. Otherwise, the Time Ratio (TR) is one. Dimensionless.

4.72E-4 = The conversion factor in $min*m^3/sec\text{-}ft^3$.

6.8 Dose Rate Calculations

Compliance with the dose rate limits for noble gases in the Specification in Step 6.14.6 is demonstrated by setting each gaseous effluent monitor alarm setpoint so that an alarm will occur at or before either dose rate limit Specification in Step 6.14.6 is reached. In addition, the Specification in Step 6.14.6 provides a maximum limit on organ dose rate equivalent beyond the Site Area Boundary from iodine-131, iodine-133, tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days. Compliance is determined by calculating the organ dose rate for the MAXIMUM EXPOSED INDIVIDUAL for the inhalation pathway only.

The dose rate due to noble gas is evaluated as follows:

Total Body:

$$\dot{D}_{tb} = (X/Q)\ \Sigma_v\ \Sigma_i (Q_{vi} * K_i)$$

Skin:

$$\dot{D}_s = (X/Q)\ \Sigma_v\ \Sigma_i\ Q_{vi} * (L_i + 1.1\ M_i)$$

where:

$\dot{D}_{tb}$ = The total body dose rate from noble gases (mrem/yr)

$\dot{D}_s$ = The skin dose rate from noble gases (mrem/yr)

X/Q = The most conservative atmospheric dispersion factor for a ground level release to a sector at or beyond the Site Area Boundary (sec/m^3). Either an annual average, multi-year average, or concurrent meteorology is acceptable. Normally taken from Attachment 4.

Q_{vi} = The release rate of noble gas radionuclide i from effluent vent v during the time of the release (μCi/sec)

K_i = A factor converting time integrated, ground-level concentration of noble gas radionuclide i to total body dose from its gamma radiation (mrem-m^3/µCi-yr). See Attachment 12.

L_i = A factor converting gamma radiation from noble gas radionuclide i to skin dose (mrem-m^3/µCi-yr). See Attachment 12.

M_i = A factor converting gamma radiation from noble gas radionuclide i to air dose (mrad-m^3/µCi-yr). See Attachment 12.

1.1 = A factor converting air dose from gamma radiation to skin dose equivalent (mrem/mrad)

The organ dose rate resulting from inhalation is calculated with the equation:

Organ:

$$\dot{DO}_{aj} = (X/Q)\ \Sigma_v\ \Sigma_i(Q_{vi} * R_{aji})$$

Where:

$\dot{DO}_{aj}$ = The dose commitment rate to organ j of a person in age group a (mrem/yr)

R_{aji} = The factor to convert air concentration of radionuclide i to organ j dose commitment rate of a person in age group a exposed by inhalation (mrem-m^3/µCi-yr). See Attachment 12.

Q_{vi} = The release rate of radionuclide i (not including Noble Gas nuclides), via effluent vent v during the time of the release (µCi/sec)

Exposure to dose rate factors, R_{aji}, for inhalation are derived by using equation 13 in RG 1.109, Rev. 1. Tables E-5, E-7, E-8, E-9, and E-10 are assumed to represent the Maximum Exposed Individual in the equation to derive R_{aji}.

6.9 Air Dose Calculations

The Surveillance Requirement in Step 6.14.7 requires cumulative dose to air from radioactive effluent noble gases to be determined in order to assess compliance with the Specification in Step 6.14.7. The air dose is evaluated in the sector of the maximum exposure at or beyond the Site Boundary for Gaseous Effluent.

Air dose from noble gas gamma radiation is calculated cumulatively with the equation:

$$D_g = 3.17E\text{-}8 \ \Sigma_v \ (X/Q) \ \Sigma_i \ \Sigma_n \ (Q_{vni} * M_i)$$

Air dose from noble gas beta radiation is calculated cumulatively with the equation:

$$D_b = 3.17E\text{-}8 \ \Sigma_v \ (X/Q) \ \Sigma_i \ \Sigma_n \ (Q_{vni} * M_i)$$

Where:

D_g = The noble gas gamma dose to air (mrad)

D_b = The noble gas beta dose to air (mrad)

X/Q = The atmospheric dispersion factor for noble gas effluent from a ground-level release to a sector beyond the Site Boundary for Gaseous Effluents (sec/m^3). Either an annual average, multi-year average, or concurrent meteorology is acceptable.

M_i = A factor converting ground-level concentration to gamma radiation from noble gas radionuclide i to air dose (mrad-m^3/µCi-yr)

N_i = A factor converting ground-level concentration to beta radiation from noble gas radionuclide i to air dose (mrad-m^3/µCi-yr)

Q_{vni} = The quantity of each noble gas radionuclide i in batch n released via effluent stream v (µCi)

$3.17 \times 10^{-8} = 1$ yr/3.156×10^7 sec

Factors M_i and N_i are 10^6 pCi/µCi times the values in RG 1.109, Rev. 1, Table B-1, Columns 4 and 2, respectively. The computer codes GASPAR and REIMS may be used to perform these calculations.

6.10 Organ Dose Calculations for Gaseous Effluents

The Surveillance Requirement in Step 6.14.8 requires the radiation dose or dose commitment to the Maximum Exposed (Hypothetical) Individual accumulated from exposure to tritium, iodine-131, iodine-133, and radioactive materials in particulate form having half-lives greater than 8.0 days, that originate in effluent air, be determined at least every month. The radiation dose or dose commitment accumulated during a calendar quarter and a year may not exceed values stated in the Specification in Step 6.14.8.

A person may be exposed to effluent radioactive material of this type in air by inhalation or indirectly via environmental pathways that involve deposition onto vegetation and the ground. The exposure pathways evaluated will include the following:

P	Exposure Pathway
1	Air--inhalation
2	Deposition onto ground-irradiation
3	Deposition onto vegetation--ingested
4	Deposition onto forage--cow--milk--ingestion
5	Deposition onto forage--meat animal--meat--ingestion
6	Deposition onto forage--goat--milk--ingestion

The equation used to calculate the dose commitment to the Maximum Exposed (Hypothetical) Individual from radionuclides other than tritium is:

$$D_{aj} = (X/Q)_{p=1} * \Sigma_v \Sigma_i (Q_{vi} * R_{ajip=1}) + {}_{p=2}\Sigma (D/Q)_p \Sigma_v \Sigma_i (Q_{vi} * R_{aijp})$$

Where: p = 1, i.e., air-inhalation, in the first term, and p = 2, 3, 4, 5, and 6 in the second term of the equation

i excludes H-3

D_{aj} = The dose commitment to organ j of a person in age group a (mrem)

Q_{vi} = The quantity of tritium, radioiodines, or radionuclides in particulate form having half-lives greater than 8.0 days in air discharged via effluent stream v (µCi)

X/Q = The atmospheric dispersion factor for a ground-level or building wake discharge. Either an annual average, multi-year average, or concurrent meteorology is acceptable (sec/m^3).

D/Q = Relative deposition factor. i.e., Factor converting a ground-level or building wake discharge in air to a real deposition on land (m^{-2}). Either an annual average, multi-year average, or concurrent meteorology is acceptable. The locations used for calculations should be from the most recent land use census data.

R_{ajip} = A factor converting time integrated concentration of radionuclide i in air or deposited on vegetation and/or ground to radiation dose commitment to organ j, including total body, of a person in age group a who is exposed via pathway p.

When p=1, representing air-inhalation, R_{ajip} has units of mrem-m^3/μCi-yr. When p=2,3,4,5 or 6 in the second term of the equation above, representing pathways involving deposition, R_{ajip} has units of mrem-m^2-sec/yr-μCi. When the radionuclide is H-3, R_{ajip} has units of mrem-m^3/μCi-yr.

Tritium is assumed not to deposit onto vegetation or the ground. Hence, the concentration in vegetation is assumed to be related to the local atmospheric concentration as described in RG 1.109, Rev. 1, Appendix C. The dose commitment to the Maximum Exposed (Hypothetical) Individual from tritium in gaseous effluent is calculated with the equation:

$$D_{aj} = 3.17\times10^{-8} \quad \Sigma\, X/Q_p \quad \Sigma(Q_{vi} * R_{ajip})_{i=H-3}$$

Where:

p = 1, 3, 4, 5, and 6

i includes H-3 only

X/Q = The atmospheric dispersion factor for a ground-level or building wake discharge. Either an annual average, multi-year average, or concurrent meteorology is acceptable (sec/m^3).

3.17×10^{-8} = years/sec

Other terms as defined above.

Dose factors R_{aijp} for RSNGS are derived using the equations and methods in RG 1.109, Rev. 1, Appendix C. Values of parameters in RG 1.109, Rev. 1, Table E-5 are assumed to represent the Maximum Exposed (Hypothetical) Individual unless Land Use Census data justify a different value. Any different values from default values will be justified and added as a table to the ODCM. Values of other parameters recommended in RG 1.109, Rev. 1, including those recommended in the absence of site-specific data, are used in the equations to derive the dose factors. (GASPAR or REIMS may be used to perform the calculations.)

6.11 Gas Dose Projections

31-Day Dose projections are calculated to show compliance with the Specification in Step 6.14.9. 31-day dose projections do not need to be performed when GASEOUS EFFLUENT RADWASTE TREATMENT SYSTEMS are being fully utilized. These systems are considered operable by meeting the Specifications in Steps 6.14.6, 6.14.7, and 6.14.8. The dose projection equations are the same as used for liquid per Step 6.5.

6.12 Fuel Cycle Dose

If a calculated dose exceeds twice the limit of the Specification in Step 6.14.3, 6.14.7, 6.14.8 or a level in Table 3 of the REMP Manual is exceeded, an assessment of compliance with the Specification in Step 6.14.10 must be made.

Liquid dose calculations shall be made using the general methodology of Step 6.4. Gas dose calculations shall be made using the general methodology of Steps 6.9 and 6.10. These methodologies are to be used as a guide and strict adherence is not required because the Fuel Cycle Dose Calculation is done to determine the actual dose received, not a hypothetical maximum. Therefore, parameters such as dilution beyond the site boundary and residential shielding may be factored into the calculation.

The total body and organ doses shall be the result of summing the individual contributions from liquid, gas, and direct radiation sources for the affected Member of the Public.

Irradiation, i.e., exposure to an external source of radiation, directly from the RSNGS normally will be evaluated with the aid of environmental dosimetry such as thermoluminescent dosimetry (TLD) or portable ion chamber (PIC) measurements.

6.13 EPA Reporting Requirements (CCTS T890828001)

If a calculated dose exceeds the Specification limit of Step 6.14.2, 6.14.3, 6.14.6, 6.14.7, or 6.14.8, an assessment of compliance with 40 CFR Parts 302 and 355, Reportable Quantity Adjustment - Radionuclides, must be made.

This involves determining the maximum quantity of radionuclides released in a 24 hour period and comparing the quantities to the values listed in 10 CFR 302 Appendix B. The "sum of the ratios" method shall be used to determine compliance. If the "sum of the ratios" is greater than one, the National Response Center shall be notified.

Since Rancho Seco's systems and procedures are set up to normally operate within the above limits, this condition is not expected to occur, therefore, specific implementation procedures to determine compliance are not required.

6.14 Technical Requirements

6.14.1 Liquid Effluent Monitoring Instrumentation

Specifications:

The radioactive liquid effluent monitoring instrumentation channels shown in Attachment 14 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Step 6.14.2 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with Step 6.2.

Applicability:

During releases via the retention basin effluent discharge.

Action:

1) With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Step 6.14.2 are met, immediately suspend the release of radioactive effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

2) With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown on Attachment 14.

Surveillance Requirements:

1) The maximum setpoint shall be determined in accordance with methodology as described in Step 6.2 and shall be recorded on the release permits.

2) 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 Attachment 15.

3) Records shall be maintained in accordance with the Process Standards of all radioactive liquid effluent monitoring instrumentation alarm/trip setpoints. Maximum setpoints and calculations shall be available for review to ensure that the limits of Step 6.14.2 are met.

6.14.2 Maximum Permissible Concentrations in Liquid Effluents

Specifications:

The concentration of radioactive material released in liquid effluents at any time beyond the Boundary for Liquid Effluents (see Attachment 5) shall be limited to the concentrations specified in 10 CFR 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} µCi/ml total activity.

Applicability:

This is applicable at all times.

Action:

With the concentration of radioactive material released from the site exceeding the above Specifications, immediately restore concentration within the required limits and report the event in the next Semi-Annual Radioactive Effluent Release Report.

Surveillance Requirements:

The concentration of radioactive material at any time in liquid effluents released from the site shall be continuously monitored in accordance with Attachment 14.

The liquid effluent continuous monitor having provisions for automatic termination of liquid releases, as listed in Attachment 14, shall be used to limit the concentration of radioactive material released at any time from the site to areas beyond the site boundary to the limits given in the above Specifications.

The radioactivity concentration of each Retention Basin to be discharged shall be determined prior to release by sampling and analysis in accordance with Attachment 16, Item A. The results of Retention Basin pre-release sample analyses shall be used with the calculational methods described below to ensure that the concentration at the point of release is within the limits of the above Specification.

6.14.3 Liquid Dose Calculations

Specifications:

The dose or dose commitment to a MAXIMUM EXPOSED (HYPOTHETICAL) INDIVIDUAL from radioactive materials in liquid effluents released beyond the Boundary for Liquid Effluents (see Attachment 6) shall be limited to:

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

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

Applicability:

At all times.

Action:

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

Surveillance Requirements:

Cumulative dose assessments associated with the release of radioactive liquid effluent shall be determined by sampling and analysis in accordance with Attachment 16, Item B and calculations performed in accordance with the methodology described in Step 6.4 at the following frequencies:

1) Prior to the initiation of a release of radioactive liquid effluent from the A or B RHUT; and,

2) Upon verification of monthly composite analysis results for radioactive liquid effluent released from the A and B RHUTs.

A dose tracking system and administrative dose limits shall be established and maintained. With the 31-day dose projection in excess of the limits in Step 6.14.4 Specifications, adjust liquid effluent operating parameters to give reasonable assurance of compliance with the dose limits of the Specification in Step 6.14.3 (10 CFR 50, Appendix I dose guidelines) and maintain radioactive liquid releases as low as is reasonably achievable.

6.14.4 Liquid Effluent Radwaste Treatment

Specifications:

The LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the system shall be used to reduce the quantity of radioactive materials in liquid effluents prior to their discharge when projected doses due to the liquid effluent beyond the Boundary for Liquid Effluents (see Attachment 6), when averaged over 31 days, would exceed 0.25 mrem to the total body or 0.83 mrem to any organ.

Applicability:

At all times.

Action:

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

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

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

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

Surveillance Requirements:

Doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days in accordance with the methodology described in Step 6.5 when LIQUID EFFLUENT RADWASTE TREATMENT SYSTEMS are not being fully utilized. The installed LIQUID EFFLUENT RADWASTE TREATMENT SYSTEM shall be considered OPERABLE by meeting the Specifications in Steps 6.14.2 and 6.14.3.

6.14.5 Gaseous Effluent Monitoring Instrumentation

Specifications:

The radioactive gaseous effluent monitoring instrumentation channels shown in Attachment 17 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Step 6.14.6 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the methodology contained in the following procedures. Continuous samples of the gaseous effluent for radioiodines and radioactive particulate material shall be taken as indicated in Attachment 17.

Applicability:

This is applicable at all times.

Action:

1) With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure that the limits of Step 6.14.6 are met, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

2) With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, taken the ACTION shown in Attachment 17. Exert best efforts to return the instrument to OPERABLE status within 30 days and if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

Surveillance Requirements:

The maximum setpoints shall be determined by procedures implementing the methodology presented in this procedure and shall be recorded on release permits.

Each 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 Attachment 18.

Records shall be maintained in accordance with the Process Standards of all radioactive gaseous effluent monitoring instrument alarm/trip setpoints. Maximum setpoints and setpoint calculations shall be available for review to ensure that the limits of Step 6.14.6 are met.

6.14.6 Gaseous Dose Rates

Specifications:

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at or beyond the Site Boundary for Gaseous Effluents (see Attachment 5) shall be limited to the following values:

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

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

Applicability:

This is applicable at all times.

Action:

With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the limit(s) specified and report the event in the next Semi-Annual Radioactive Effluent Release Report.

Surveillance Requirements:

The noble gas effluent continuous monitors, as listed in Attachment 17, shall use monitor setpoints to limit the dose rate in unrestricted areas to the limits in the above Specification.

In the event a noble gas effluent exceeds the setpoint of its monitor, an assessment of compliance with the Specification above shall be made in accordance with the methodology contained in this manual.

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 analyses program, specified in Attachment 18.

The dose rate due to Iodine-131, Iodine-133, tritium, and all radioactive material in particulate form with half-lives greater than 8 days, released in gaseous effluents, shall be determined to be within the limits in the above Specification by using the results of the sampling and analysis program specified in Attachment 19, and in accordance with the methodology described in Step 6.8.

6.14.7 Gamma and Beta Air Dose

Specifications:

The air dose due to noble gases released in gaseous effluents to areas at or beyond the Site Boundary for Gaseous Effluents (see Attachment 5) shall be limited to the following:

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

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

Applicability:

This is applicable at all times.

Action:

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

Surveillance Requirements:

Cumulative air dose contributions for the calendar quarter and calendar year shall be determined in accordance with the methodology in Step 6.9 at least monthly.

6.14.8 Gaseous Organ Dose

Specifications:

The dose or dose commitment to a MAXIMUM EXPOSED (HYPOTHETICAL) INDIVIDUAL from Iodine-131, Iodine-133, tritium, and radioactive materials in particulate form with half-lives greater than eight days in gaseous effluents released to areas at or beyond the Site Boundary for Gaseous Effluents (see Attachment 5) shall be limited to the following:

1) During any calendar quarter, to less than or equal to 7.5 mrem to any organ; and,

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

Applicability:

This is applicable at all times.

Action:

With the calculated dose or dose commitment from the release of Iodine-131, Iodine-133, tritium, and radioactive materials in particulate form with half-lives greater than eight days in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report. This Report will identify the cause(s) for exceeding the limit and define the corrective actions to be taken to reduce the releases and the proposed corrective action(s) to be taken to assure that subsequent releases will be in compliance with the above annual limits.

Surveillance Requirements:

Cumulative dose contributions for the calendar quarter and calendar year period shall be determined in accordance with the methodology described in Step 6.10 at least monthly.

6.14.9 Gaseous Dose Projections

Specifications:

The VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of this system shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected air doses due to gaseous effluent releases (see Attachment 5), averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the Ventilation Exhaust Treatment System shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Attachment 5) when averaged over 31 days, would exceed 0.3 mrem to any organ.

Applicability:

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

Action:

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

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

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

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

Surveillance Requirements:

Doses due to gaseous releases to areas at and beyond the Site Boundary for Gaseous Effluents (see Attachment 5) shall be projected at least once per 31 days in accordance with the methodology and parameters in Step 6.11 when the Gaseous Radwaste Treatment Systems are not being fully utilized.

The installed VENTILATION EXHAUST TREATMENT SYSTEM shall be considered OPERABLE by meeting the Specifications in Steps 6.14.6, 6.14.7, and 6.14.8.

6.14.10 Fuel Cycle Dose

Specification

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

Applicability At all times

Action

1) With the calculated doses from the release of radioactive material in liquid or gaseous effluents exceeding twice the limits of Specifications in Steps 6.14.3, 6.14.7, 6.14.8 or exceeding the reporting levels in Table 3 of the REMP Manual, calculations shall be made including direct radiation contributions (including outside storage tanks, etc.) to determine whether the above specifications have been exceeded.

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

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

6.14.11 Air Filter Systems

Surveillance Requirements

DOP tests will be performed on the HEPA filters in the Ventilation Exhaust Treatment Systems every 18 months, or after any work has been performed on the HEPA filter systems which could alter their integrity.

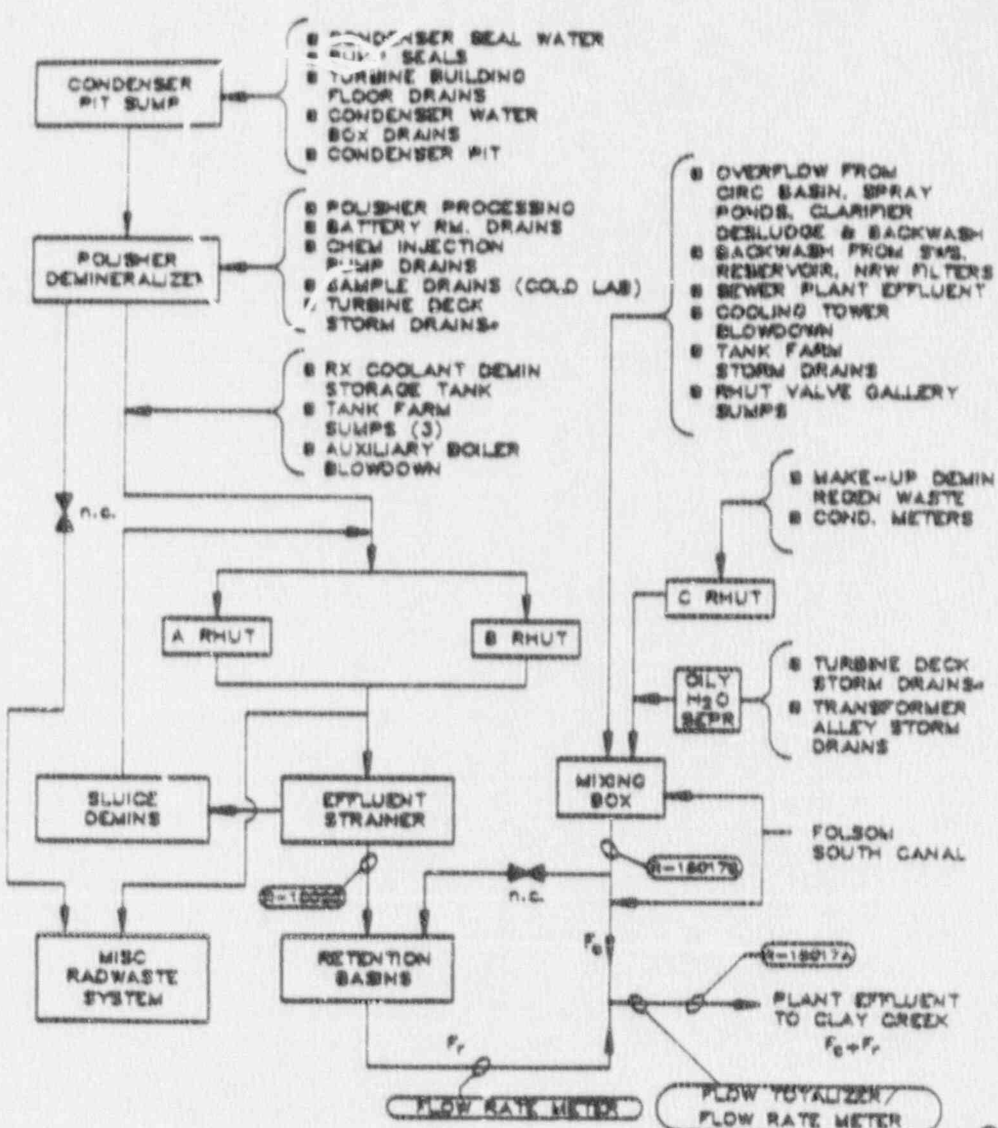
7.0 RECORDS

The following individual/packaged documents and related correspondence completed as a result of the performance or implementation of this procedure are records. They shall be transmitted to Records Management in accordance with RSAP-0601, Nuclear Records Management.

None

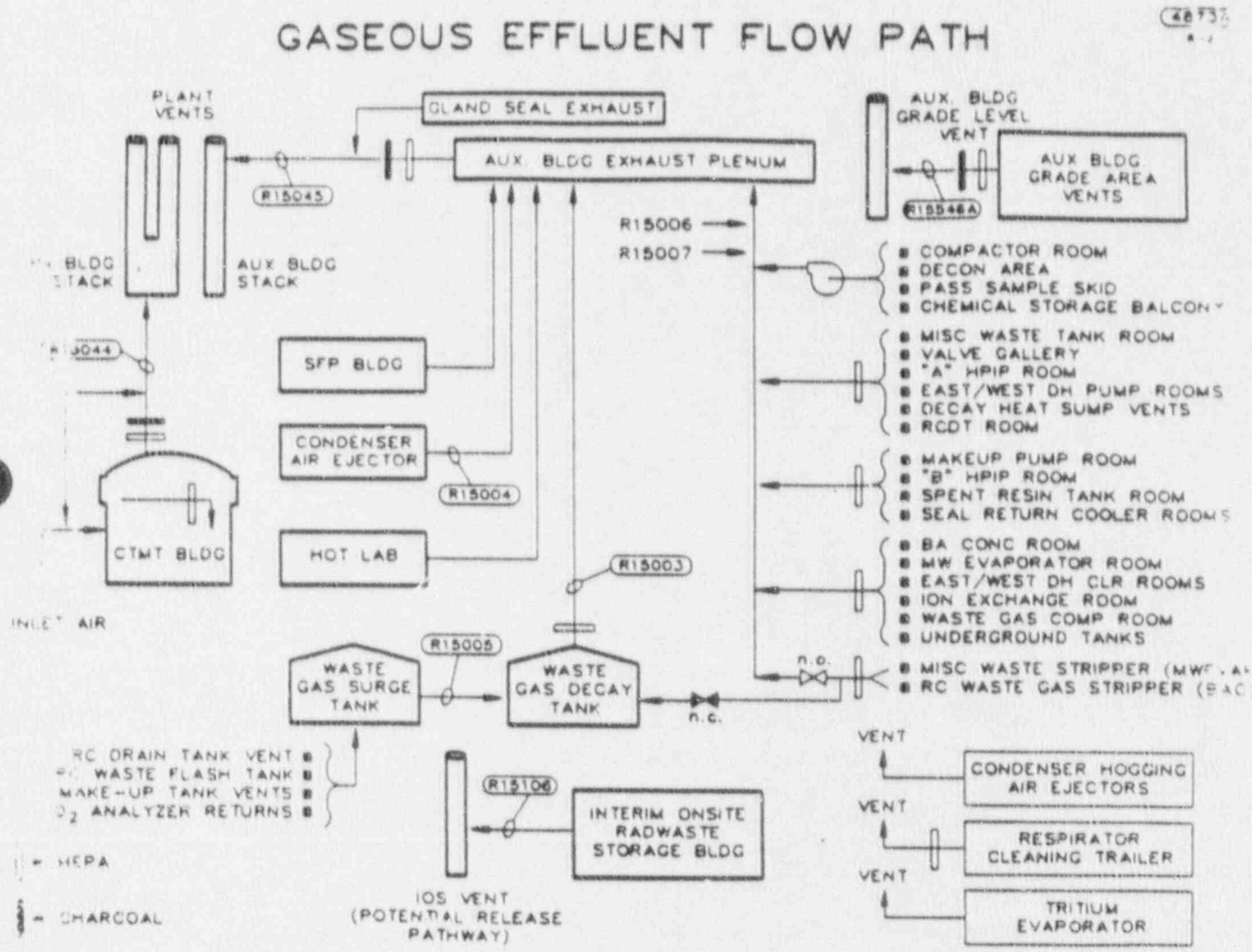
LIQUID EFFLUENT FLOW DIAGRAM

LIQUID EFFLUENT FLOW DIAGRAM



* TURBINE DECK STORM DRAINS CAN BE DIRECTED TO THE OILY WATER SEPARATOR (NORMAL) OR THE POLISHER DEMIN SUMP

GASEOUS EFFLUENT FLOW DIAGRAM



DOSE FACTOR PARAMETERS

CONSUMPTION and USAGE PARAMETERS

PATHWAY	AGE	1.109 DEFAULT	RSNGS
Irrigated Stored Vegetables	Adult	520 kg/yr	520
	Teen	630 kg/yr	630
	Child	520 kg/yr	520
	Infant	0 kg/yr	0
Irrigated Fresh Vegetables	Adult	64 kg/yr	64
	Teen	42 kg/yr	42
	Child	26 kg/yr	26
	Infant	0 kg/yr	0
Irrigated Milk	Adult	310 kg/yr	310
	Teen	400 kg/yr	400
	Child	330 kg/yr	330
	Infant	330 kg/yr	330
Irrigated Meat & Poultry	Adult	110 kg/yr	110
	Teen	65 kg/yr	65
	Child	41 kg/yr	41
	Infant	0 kg/yr	0
Fish	Adult	21 kg/yr	21
	Teen	16 kg/yr	16
	Child	6.9 kg/yr	6.9
	Infant	0 kg/yr	0
Other Seafood Invertebrate (Crayfish)	Adult	5.0 kg/yr	6.9
	Teen	3.8 kg/yr	5.2
	Child	1.7 kg/yr	2.2
	Infant	0 kg/yr	0
Algae	Adult	None	0
	Teen	None	0
	Child	None	0
	Infant	None	0
Water usage (Drinking Water)	Adult	730 l/yr	0
	Teen	510 l/yr	0
	Child	510 l/yr	0
	Infant	330 l/yr	0

DOSE FACTOR PARAMETERS (Continued)

CONSUMPTION and USAGE PARAMETERS (Continued)

PATHWAY	AGE	1.109 DEFAULT	RSNGS
Shoreline Recreation	Adult	12 hr/yr	200
	Teen	67 hr/yr	100
	Child	14 hr/yr	14
	Infant	0 hr/yr	0
Swimming	Adult	None	100
	Teen	None	100
	Child	None	14
	Infant	None	0
Boating	Adult	None	0
	Teen	None	0
	Child	None	0
	Infant	None	0
Inhalation	Adult	8000 m^3	8000
	Teen	8000 m^3	8000
	Child	3700 m^3	3700
	Infant	1400 m^3	1400

IRRIGATION RATES AND FRACTION OF IRRIGATION

	1.109 DEFAULT	RSNGS
Irrigation Rate	263 $l/m_2/mo$	263
Time field has been irrigated prior to crop of interest	15 years	15
Fraction of year field is irrigated	None	None
Fraction of animal water intake not obtained from the irrigation system (Irrigated Meat)	None	0
Fraction of animal water intake not obtained from the irrigation system (Irrigated Milk)	None	1

DOSE FACTOR PARAMETERS (Continued)

TRANSIT, TRANSFER and HOLDUP TIMES

	1.109 DEFAULT	RSNGS
Irrigated Stored Vegetables Holdup Time	1440 hrs	1440
Irrigated Fresh Vegetables Holdup Time	24 hrs	24
Irrigated Milk Holdup Time	48 hrs	48
Irrigated Meat Holdup Time	480 hrs	480
Transit Time from time of sample to time of release	None	72 hrs
Transit Time to drinking water	None	0

DILUTIONS

	1.109 DEFAULT	RSNGS
All pathways	None	1 (None)
Shore-width factor	0.2	0.2

MISCELLANEOUS

	1.109 DEFAULT	RSNGS
Fraction of Leafy Vegetables grown in garden of interest	1.0	1.0
Fraction of produce ingested grown in garden of interest	.76	.76
Crop Growing Time for Leafy Vegetables ingested by man	60 days	30
Crop Growing Time for Pasture Grass	30 days	30
Grop Yield for Leafy Vegetables ingested by man	2.0 kg/m^2	2.0
Crop Yield for Pasture Grass	0.7 kg/m^2	2.0

ATMOSPHERIC DISPERSION AND DEPOSITION PARAMETERS
GASEOUS EFFLUENT PATHWAYS
1989 CONTROLLING LOCATIONS**

PATHWAY	DIRECTION	DISTANCE	X/Q (m/sec^2)	D/Q* (m^{-2})
Inhalation	ENE	1128 M	7.2E-06	--
Ground	ENE	1128 M	--	4.1E-08
Vegetation	ENE	1128 M	7.2E-06	4.1E-08
Cow Milk	ENE	1128 M	7.2E-06	4.1E-08
Meat Animal	S	195 M	1.2E-04	--
Meat Animal	SSE	198 M	--	3.4E-07
Site Boundary	NNW	670 M	2.1E-05	--

ATMOSPHERIC DISPERSION AND DEPOSITION PARAMETERS
GASEOUS EFFLUENT PATHWAYS FOR THE IOS
1989 CONTROLLING LOCATIONS**

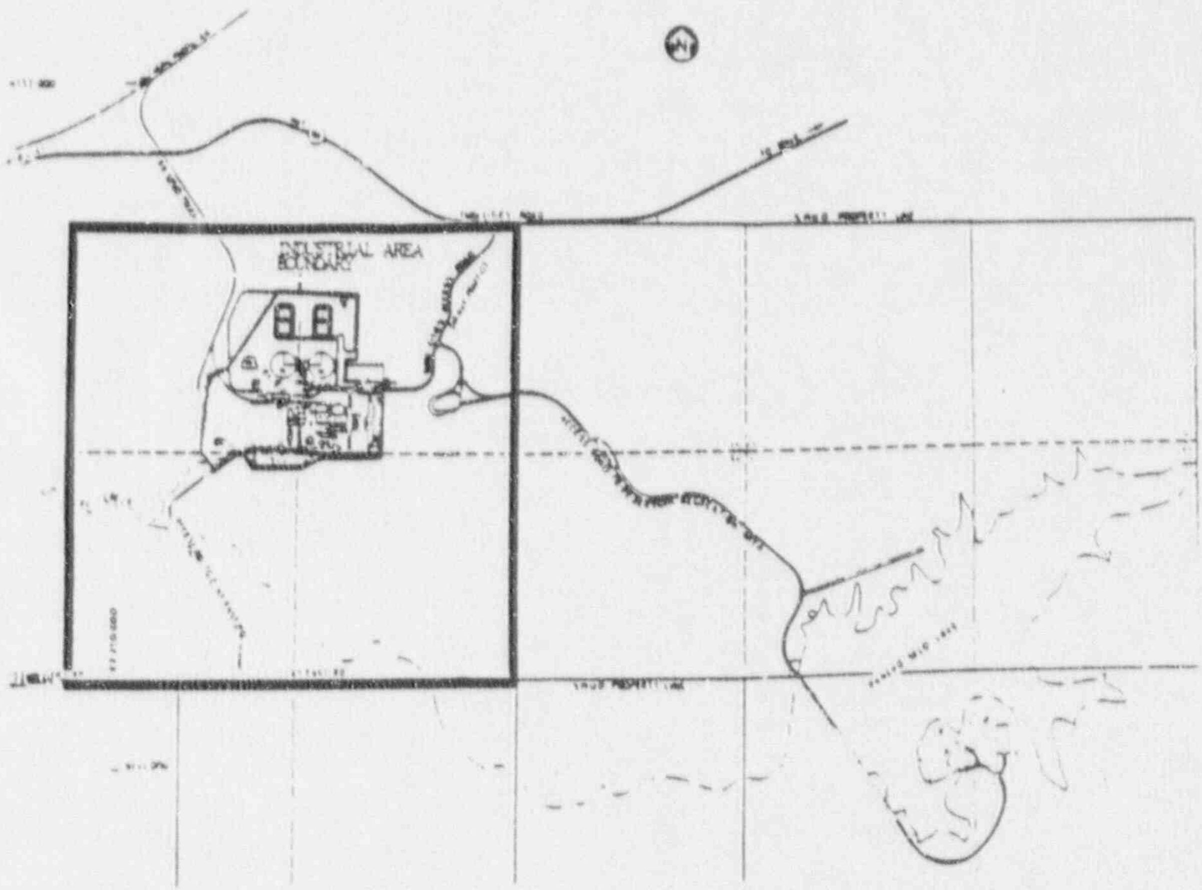
PATHWAY	DIRECTION	DISTANCE	X/Q (m/sec^2)	D/Q* (m^{-2})
Inhalation	ENE	1402 M	4.5E-06	--
Ground	ENE	1402 M	--	2.5E-08
Vegetation	ENE	1402 M	4.5E-06	2.5E-08
Cow Milk	ENE	1402 M	4.5E-06	2.5E-08
Meat Animal	WNW	80 M	8.2E-04	1.3E-06
Site Boundary	NNW	488 M	3.5E-05	--

* Based on meteorological data from January 1977 to December 1987.

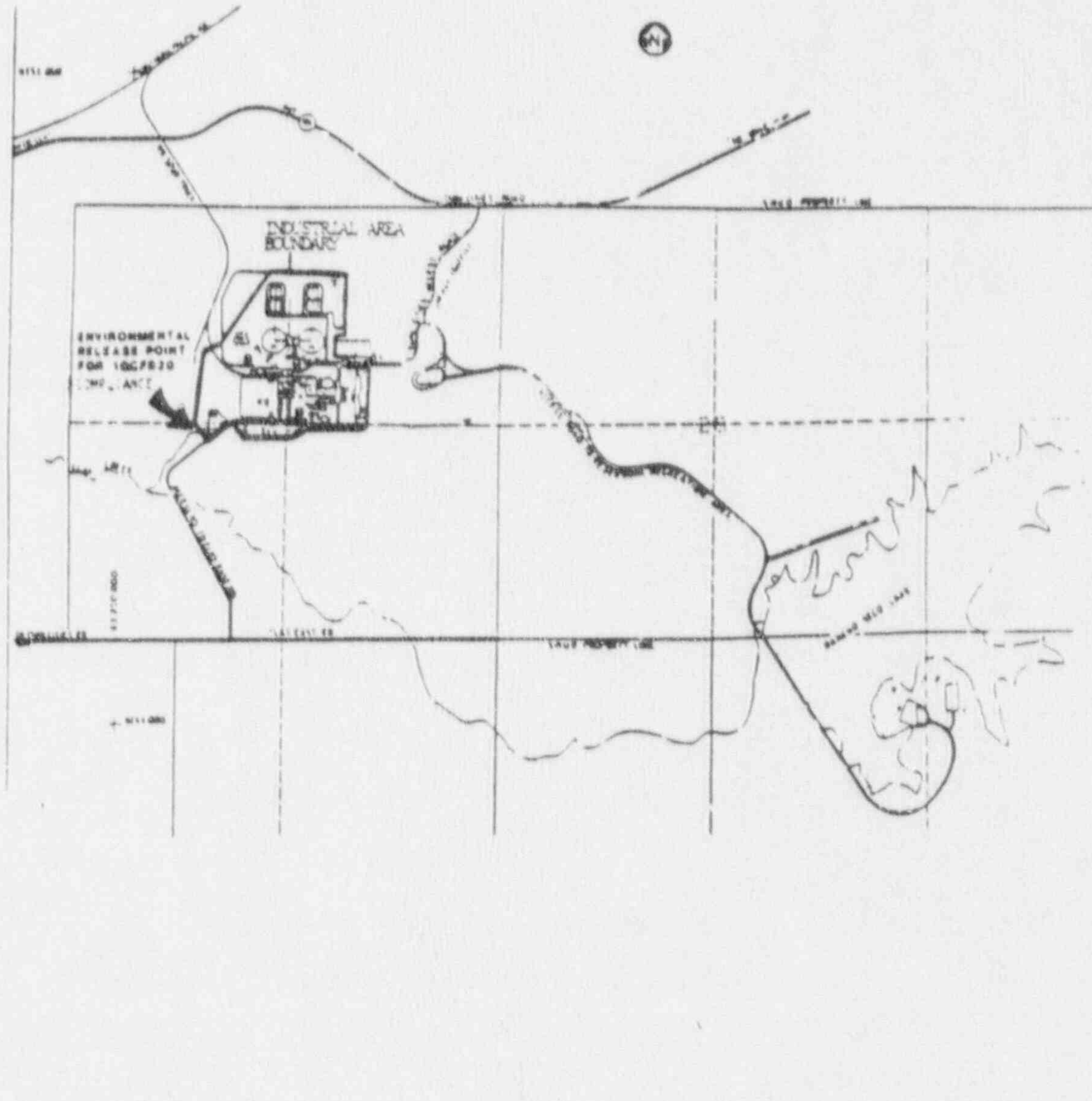
** Based on 1989 Land Use Census.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

For compliance With 10 CFR 20 and 10 CFR 50



BOUNDARY FOR LIQUID EFFLUENTS



HISTORICAL LIQUID SOURCE TERMS*

		Gamma Emitters Relative		All Nuclides Relative	
Nuclear	C_1 (µCi/ml)	%	fraction	%	fraction
CO-60	3.34E-09	14.02	0.1402	.33	0.0033
CS-134	2.09E-09	8.77	0.0877	.21	0.0021
CS-137	1.84E-08	77.21	0.7721	1.82	0.0182
H-3	9.88E-07	0	0	97.64	0.9764
TOTAL	1.01E-06	100.00	1.0	100.00	1.0

* Based on Retention Basin Effluent History of Rancho Seco (July 1 - December 1, 1989)

LIQUID MONITOR DETECTOR EFFICIENCIES

R15017A & R15017B*
(Detector Model RD-53)

Nuclide	Efficiency cpm/µCi/cc
MO-99	3.47E+07
I-131	1.90E+08
I-132	4.17E+08
I-133	1.53E+08
I-135	1.74E+08
CS-134	3.25E+08
CS-137	1.28E+08
CR-51	1.85E+07
MN-54	1.30E+08
FE-59	1.26E+08
CO-58	1.85E+08
CO-60	2.40E+08
ZN-65	6.49E+07
SB-124	2.66E+08
BA-140	8.98E+07
LA-140	2.69E+08
CE-141	7.68E+07
CE-144	1.80E+07

* From Calculation Z-RDM-I0261

ORGAN DOSE FACTORS FOR LIQUID EFFLUENTS

DOSE FACTOR TABLE: A(i) - Adult

Units are mrem/hr per uCi/ml

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	1.210E+01	1.210E+01	1.210E+01	1.210E+01	1.210E+01	1.210E+01
C-14	3.820E+05	7.630E+04	7.630E+04	7.630E+04	7.630E+04	7.630E+04	7.630E+04
NA-24	4.320E+01	4.320E+01	4.320E+01	4.320E+01	4.320E+01	4.320E+01	4.320E+01
P-32	4.070E+07	2.530E+06	1.570E+06	6.310E-02	6.310E-02	6.310E-02	4.580E+06
CR-51	3.080E+00	3.080E+00	8.890E+00	6.560E+00	4.360E+00	1.080E+01	1.460E+03
MN-54	8.230E+02	3.310E+05	6.380E+04	8.230E+02	9.900E+04	8.230E+02	1.010E+06
MN-56	0.000E+00	2.130E-04	3.790E-05	0.000E+00	2.700E-04	0.000E+00	6.790E-03
FE-55	1.180E+04	8.170E+03	1.910E+03	7.290E-04	7.290E-04	4.560E+03	4.690E+03
FE-59	1.470E+04	3.420E+04	1.320E+04	1.770E+02	1.770E+02	9.690E+03	1.140E+05
CO-58	2.360E+02	9.910E+02	1.930E+03	2.360E+02	2.360E+03	2.360E+02	1.550E+04
CO-60	1.260E+04	1.610E+04	2.020E+04	1.260E+04	1.260E+04	1.260E+04	7.730E+04
NI-63	3.420E+05	2.370E+04	1.150E+04	0.000E+00	0.000E+00	0.000E+00	4.950E+03
NI-65	8.650E-04	1.120E-04	5.130E-05	0.000E+00	0.000E+00	0.000E+00	2.850E-03
CU-64	6.900E-02	7.020E-01	3.770E-01	8.900E-02	1.640E+00	8.900E-02	5.240E+01
ZN-65	7.630E+04	2.420E+05	1.100E+05	4.460E+02	1.620E+05	4.460E+02	1.520E+05
ZN-69	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-83	0.000E+00	0.000E+00	4.840E-05	0.000E+00	0.000E+00	0.000E+00	6.980E-05
BR-84	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-86	6.460E+00	1.210E+05	5.620E+04	6.460E+00	6.460E+00	6.460E+00	2.380E+04
RB-88	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-89	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-89	2.010E+05	6.250E-02	5.750E+03	6.250E-02	6.250E-02	6.250E-02	3.220E+04
SR-90	5.710E+06	6.160E-03	1.530E+06	6.160E-03	6.160E-03	6.160E-03	3.260E+05
SR-91	5.590E+00	1.190E-01	3.400E-01	1.190E-01	1.190E-01	1.190E-01	2.620E+01
SR-92	5.950E-03	0.000E+00	2.580E-04	0.000E+00	0.000E+00	0.000E+00	1.180E-01
Y-90	3.150E+00	6.940E-02	1.520E-01	6.940E-02	6.940E-02	6.940E-02	3.270E+04
Y-91M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y-91	1.950E+02	6.820E-01	5.890E+00	6.820E-01	6.820E-01	6.820E-01	1.070E+05
Y-92	1.720E-05	4.240E-06	4.620E-06	4.240E-06	4.240E-06	4.240E-06	2.280E-01
Y-93	2.160E-02	1.610E-02	1.630E-02	1.610E-02	1.610E-02	1.610E-02	1.730E+02
ZR-95	1.820E+02	1.640E+02	1.620E+02	1.560E+02	1.690E+02	1.560E+02	2.670E+04
ZR-97	9.870E-01	9.830E-01	9.820E-01	9.820E-01	9.830E-01	9.820E-01	3.260E+02
NB-95	5.160E+02	3.270E+02	2.170E+02	8.920E+01	3.240E+02	8.920E+01	1.440E+06
MO-99	1.270E+00	3.040E+02	5.890E+01	1.270E+00	6.870E+02	1.270E+00	7.030E+02
TC-99M	8.230E-04	1.030E-03	[illegible]	7.110E-04	5.510E-03	8.660E-04	1.880E-01
TC-101	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-103	6.760E+02	7.020E+01	3.310E+02	7.020E+01	2.380E+03	7.020E+01	7.080E+04
RU-105	9.350E-04	1.830E-04	4.800E-04	1.830E-04	9.910E-03	1.830E-04	4.600E-01
RU-106	1.610E+04	2.950E+02	2.300E+03	2.950E+02	3.080E+04	2.950E+02	1.020E+06
AG-110M	2.630E+03	2.590E+03	2.370E+03	2.060E+03	3.100E+03	2.060E+03	2.170E+05
TE-125M	1.660E+04	6.750E+03	2.490E+03	5.600E+03	7.570E+04	9.210E-01	7.430E+04
TE-127M	5.260E+04	1.880E+04	6.410E+03	1.340E+04	2.140E+05	5.570E-02	1.760E+05
TE-127	6.060E-01	2.180E-01	1.310E-01	4.490E-01	2.470E+00	1.620E-04	4.780E+01
TE-129M	7.190E+04	2.680E+04	1.140E+04	2.470E+04	3.000E+05	1.310E+01	3.620E+05
TE-129	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-131M	1.110E+03	5.440E+02	4.540E+02	8.580E+02	5.450E+03	5.640E+00	5.340E+04
TE-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-132	6.300E+03	4.070E+03	3.820E+03	4.500E+03	3.920E+04	3.730E+00	1.930E+05
I-130	3.140E+00	7.620E+00	3.510E+00	5.760E+02	1.140E+01	8.370E-01	6.680E+00
I-131	7.060E+02	1.000E+03	5.820E+02	3.240E+05	1.710E+03	1.470E+01	2.760E+02
I-132	1.600E-04	4.290E-04	1.500E-04	1.500E-02	6.830E-04	0.000E+00	8.050E-05
I-133	1.680E+01	2.840E+01	9.450E+00	4.010E+03	4.880E+01	1.130E+00	2.570E+01
I-134	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-135	1.590E-01	4.140E-01	1.530E-01	2.730E+01	6.630E-01	8.910E-04	4.680E-01
CS-134	4.440E+05	1.050E+06	8.590E+05	4.030E+03	3.430E+05	1.160E+05	3.230E+04
CS-136	3.160E+04	1.240E+05	8.950E+04	[illegible]	6.920E+04	9.590E+03	1.420E+04
CS-137	5.850E+05	7.980E+05	5.250E+05	6.060E+03	2.750E+05	9.540E-04	2.140E+04
CS-138	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-140	4.900E+03	2.110E+01	3.350E+02	1.500E+01	1.710E+01	1.850E+01	1.010E+04
BA-141	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	4.140E+00	3.920E+00	3.760E+00	3.710E+00	3.710E+00	3.710E+00	1.610E+04
LA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	6.290E-06
CE-141	1.870E+01	1.560E+01	9.660E+00	8.900E+00	1.200E+01	8.900E+00	2.540E+04
CE-143	1.930E+00	1.420E+02	1.760E+00	1.740E+00	1.810E+00	1.740E+00	5.240E+03
CE-144	8.990E+02	4.000E+02	8.760E+01	4.150E+01	2.540E+02	4.150E+01	2.900E+05
PR-143	7.380E+00	2.960E+00	3.660E-01	0.000E+00	1.710E+00	0.000E+00	3.230E+04
PR-144	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND-147	9.620E+00	1.030E+01	5.270E+00	4.950E+00	8.110E+00	4.950E+00	2.590E+04
W-187	2.030E+01	1.720E+01	6.890E+00	1.350E+00	1.350E+00	1.350E+00	5.200E+03
NP-239	1.700E+00	1.560E+00	1.560E+00	1.550E+00	1.590E+00	1.550E+00	2.920E+03

ORGAN DOSE FACTORS FOR LIQUID EFFLUENTS (Continued)

DOSE FACTOR TABLE: A(i) - Teen

Units are mrem/hr per uCi/ml

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	1.370E+01	1.370E+01	1.370E+01	1.370E+01	1.370E+01	1.370E+01
C-14	5.640E+05	1.130E+05	1.130E+05	1.130E+05	1.130E+05	1.130E+05	1.130E+05
NA-24	6.360E+01	6.360E+01	6.360E+01	6.360E+01	6.360E+01	6.360E+01	6.360E+01
P-32	4.440E+07	2.750E+06	1.720E+06	6.310E-02	6.310E-02	6.310E-02	3.730E+06
CR-51	1.820E+00	1.820E+00	8.100E+00	5.310E+00	3.190E+00	1.080E+01	1.060E+03
MN-54	4.200E+02	3.240E+05	6.450E+04	4.200E+02	9.690E+04	4.200E+02	6.630E+05
MN-56	0.000E+00	1.920E-04	3.430E-05	0.000E+00	2.430E-04	0.000E+00	1.260E-02
FE-55	1.370E+04	9.690E+03	2.260E+03	7.290E-04	7.290E-04	6.150E+03	4.190E+03
FE-59	1.550E+04	3.610E+04	1.400E+04	1.000E+02	1.000E+02	1.150E+04	8.530E+04
CO-58	1.280E+02	1.080E+03	2.320E+03	1.280E+02	1.280E+02	1.280E+02	1.330E+04
CO-60	6.350E+03	1.100E+04	1.690E+04	6.350E+03	6.350E+03	6.350E+03	6.710E+04
NI-63	4.500E+05	3.180E+04	1.530E+04	0.000E+00	0.000E+00	0.000E+00	5.060E+03
NI-65	8.190E-04	1.050E-04	4.770E-05	0.000E+00	0.000E+00	0.000E+00	5.670E-03
CU-64	8.550E-02	9.250E-01	4.800E-01	8.550E-02	2.210E+00	8.550E-02	6.520E+01
ZN-65	7.560E+04	2.620E+05	1.220E+05	2.290E+02	1.680E+05	2.290E+02	1.110E+05
ZN-69	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-83	0.000E+00	0.000E+00	4.930E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-84	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-86	4.100E+00	1.400E+05	6.560E+04	4.100E+00	4.100E+00	4.100E+00	2.070E+04
RB-88	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-89	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-89	2.930E+05	5.640E-02	8.380E+03	5.640E-02	5.640E-02	5.640E-02	3.480E+04
SR-90	7.400E+06	6.160E-03	1.980E+06	6.160E-03	6.160E-03	6.160E-03	3.850E+05
SR-91	5.720E+00	1.160E-01	3.390E-01	1.160E-01	1.160E-01	1.160E-01	2.560E+01
SR-92	5.550E-03	0.000E+00	2.370E-04	0.000E+00	0.000E+00	0.000E+00	1.420E-01
Y-90	3.350E+00	6.870E-02	1.570E-01	6.870E-02	6.870E-02	6.870E-02	2.710E+04
Y-91M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y-91	2.460E+02	3.780E-01	6.970E+00	3.780E-01	3.780E-01	3.780E-01	1.010E+05
Y-92	1.640E-05	4.200E-06	4.550E-06	4.200E-06	4.200E-06	4.200E-06	3.350E-01
Y-93	2.130E-02	1.570E-02	1.590E-02	1.570E-02	1.570E-02	1.570E-02	1.700E+02
ZR-95	1.190E+02	9.670E+01	9.340E+01	8.620E+01	1.020E+02	8.620E+01	2.420E+04
ZR-97	9.410E-01	9.370E-01	9.370E-01	9.360E-01	9.380E-01	9.360E-01	2.630E+02
NB-95	4.730E+02	2.820E+02	1.7[illegible]E+02	4.460E+01	2.750E+02	4.460E+01	1.020E+06
MO-99	6.360E-01	4.320E+02	8.280E+01	6.360E-01	9.870E+02	6.360E-01	7.730E+02
TC-99M	8.540E-04	1.130E-03	6.350E-03	6.980E-04	7.200E-03	9.400E-04	2.870E-01
TC-101	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-103	5.860E+02	3.510E+01	2.710E+02	3.510E+01	1.980E+03	3.510E+01	4.610E+04
RU-105	8.800E-04	1.810E-04	4.520E-04	1.810E-04	9.000E-03	1.810E-04	5.650E-01
RU-106	1.590E+04	1.480E+02	2.140E+03	1.480E+02	3.060E+04	1.480E+02	7.570E+05
AG-110M	1.870E+03	1.830E+03	1.530E+03	1.060E+03	2.530E+03	1.060E+03	2.170E+05
TE-125M	2.060E+04	7.420E+03	2.750E+03	5.750E+03	4.810E-01	4.810E-01	6.070E+04
TE-127M	5.930E+04	2.100E+04	7.050E+03	1.410E+04	2.400E+05	2.930E-02	1.480E+05
TE-127	6.520E-01	2.320E-01	1.410E-01	4.510E-01	2.640E+00	1.570E-04	5.040E+01
TE-129M	7.750E+04	2.880E+04	1.230E+04	2.500E+04	3.240E+05	7.700E+00	2.910E+05
TE-129	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-131M	1.180E+03	5.670E+02	4.740E+02	8.500E+02	5.860E+03	5.200E+00	4.510E+04
TE-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-132	6.580E+03	4.170E+03	3.920E+03	4.400E+03	4.000E+04	3.070E+00	1.320E+05
I-130	3.800E+00	9.450E+00	4.260E+00	7.060E+02	1.410E+01	8.080E-01	7.450E+00
I-131	9.720E+02	1.360E+03	7.330E+02	3.930E+05	2.330E+03	1.080E+01	2.770E+02
I-132	1.460E-04	3.810E-04	1.370E-0[illegible]	1.290E-02	6.010E-04	0.000E+00	1.660E-04
I-133	2.270E+01	3.770E+01	1.220E+01	5.120E+03	6.540E+01	1.060E+00	2.880E+01
I-134	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-135	1.850E-01	4.760E-01	1.770E-01	3.060E+01	7.520E-01	6.450E-04	5.280E-01
CS-134	5.040E+05	1.180E+06	5.500E+05	2.030E+03	3.770E+05	1.450E+05	1.670E+04
CS-136	3.270E+04	1.280E+05	8.620E+04	7.750E+01	6.990E+04	1.110E+04	1.040E+04
CS-137	6.990E+05	9.300E+05	3.260E+05	3.030E+03	3.180E+05	1.260E+05	1.620E+04
CS-138	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.940E-06
BA-140	5.340E+03	1.640E+01	3.530E+02	9.870E+00	1.210E+01	1.430E+01	8.230E+03
BA-141	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	2.300E+00	2.[illegible]40E+00	1.910E+00	1.850E+00	1.850E+00	1.850E+00	1.270E+04
LA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	[illegible].350E+05
CE-141	1.690E+01	1.300E+01	6.050E+00	5.150E+00	8.840E+00	5.150E+00	2.250E+04
CE-143	1.790E+00	1.480E+02	1.610E+00	1.590E+00	1.660E+00	1.590E+00	4.410E+03
CE-144	1.200E+03	5.100E+02	8.470E+01	2.120E+01	3.130E+02	2.120E+01	2.970E+05
PR-143	7.980E+00	3.180E+00	3.970E-01	0.000E+00	1.850E+00	0.000E+00	[illegible].620E+04
PR-144	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND-147	7.720E+00	8.170E+00	2.820E+00	2.470E+00	5.820E+00	2.470E+00	2.060E+04
W-187	2.170E+01	1.790E+01	7.110E+00	1.260E+00	1.260E+00	1.260E+00	[illegible].510E+03
NP-239	1.500E+00	1.360E+00	1.350E+00	1.340E+00	1.390E+00	1.340E+00	2.410E+03

ORGAN DOSE FACTORS FOR LIQUID EFFLUENTS (Continued)

DOSE FACTOR TABLE: A(i) - Child

Units are mrem/hr per uCi/ml

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	2.100E+01	2.100E+01	2.100E+01	2.100E+01	2.100E+01	2.100E+01
C-14	1.280E+06	2.560E+05	2.560E+05	2.560E+05	2.560E+05	2.560E+05	2.560E+05
NA-24	1.160E+02	1.160E+02	1.160E+02	1.160E+02	1.160E+02	1.160E+02	1.160E+02
P-32	5.740E+07	2.690E+06	2.210E+06	8.840E-03	8.840E-03	8.840E-03	1.590E+06
CR-51	2.540E-01	2.540E-01	7.950E+00	4.520E+00	1.420E+00	8.050E+00	4.080E+02
MN-54	5.880E+01	2.530E+05	6.740E+04	5.880E+01	7.100E+04	5.880E+01	2.120E+05
MN-56	0.000E+00	2.510E-04	5.680E-05	0.000E+00	3.040E-04	0.000E+00	3.640E-02
FE-55	2.400E+04	1.270E+04	3.940E+03	1.020E-04	1.020E-04	7.190E+03	2.350E+03
FE-59	2.210E+04	3.580E+04	1.780E+04	1.400E+01	1.400E+01	1.040E+04	3.720E+04
CO-58	1.790E+01	1.280E+03	3.880E+03	1.790E+01	1.790E+01	1.790E+01	7.380E+03
CO-60	8.890E+02	7.510E+03	2.040E+04	8.890E+02	8.890E+02	8.890E+02	3.760E+04
NI-63	1.020E+06	5.460E+04	3.480E+04	0.000E+00	0.000E+00	0.000E+00	3.690E+03
NI-65	1.520E-03	1.430E-04	8.350E-05	0.000E+00	0.000E+00	0.000E+00	1.750E-02
CU-64	1.200E-02	1.2[illegible]0E+00	7.700E-01	1.200E-02	3.050E+00	1.200E-02	5.890E+01
ZN-65	9.530E+04	2.540E+05	1.580E+05	3.210E+01	1.600E+05	3.210E+01	4.460E+04
ZN-69	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-83	0.000E+00	0.000E+00	9.580E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-84	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-86	5.740E-01	1.610E+05	9.880E+04	5.740E-01	5.740E-01	5.740E-01	1.030E+04
RB-88	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-89	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-89	6.480E+05	7.900E-03	1.850E+04	7.900E-03	7.900E-03	7.900E-03	2.510E+04
SR-90	1.470E+07	8.620E-04	3.930E+06	8.620E-04	8.620E-04	8.620E-04	2.970E+05
SR-91	1.040E+01	1.620E-02	4.060E-01	1.620E-02	1.620E-02	1.620E-02	2.280E+01
SR-92	1.020E-02	0.000E+00	4.090E-04	0.000E+00	0.000E+00	0.000E+00	1.930E-01
Y-90	4.280E+00	9.620E-03	1.240E-01	9.620E-03	9.620E-03	9.620E-03	1.220E+04
Y-91M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y-91	4.490E+02	5.290E-02	1.210E+01	5.290E-02	5.290E-02	5.290E-02	5.980E+04
Y-92	2.310E-05	0.000E+00	1.230E-06	0.000E+00	0.000E+00	0.000E+00	6.500E-01
Y-93	1.050E-02	2.200E-03	2.430E-03	2.200E-03	2.200E-03	2.200E-03	1.240E+02
ZR-95	8.320E+01	2.770E+01	2.600E+01	1.210E+01	3.450E+01	1.210E+01	1.630E+04
ZR-97	1.400E-01	1.320E-01	1.320E-01	1.310E-01	1.330E-01	1.310E-01	1.900E+02
NB-95	5.200E+02	2.060E+02	1.490E+02	6.250E+00	1.940E+02	6.250E+00	3.700E+05
MO-99	8.900E-02	7.010E+02	1.730E+02	8.900E-02	1.500E+03	8.900E-02	5.800E+02
TC-99M	4.340E-04	7.580E-04	1.100E-02	9.770E-05	9.690E-03	4.330E-04	3.760E-01
TC-101	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-103	1.020E+03	4.920E+00	3.960E+02	4.920E+00	2.570E+03	4.920E+00	2.630E+04
RU-105	1.310E-03	2.530E-05	4.900E-04	2.530E-05	1.130E-02	2.530E-05	8.350E-01
RU-106	3.180E+04	2.070E+01	3.980E+03	2.070E+01	4.290E+04	2.070E+01	4.940E+05
AG-110M	1.790E+03	1.260E+03	1.040E+03	1.480E+02	2.220E-03	1.480E+02	1.320E+05
TE-125M	3.040E+04	8.250E+03	4.060E+03	8.540E+03	6.730E-02	6.730E-02	2.940E+04
TE-127M	9.300E+04	2.500E+04	1.100E+04	2.230E+04	2.650E+05	4.100E-03	7.530E+04
TE-127	8.940E-01	2.410E-01	1.920E-01	6.190E-01	2.540E+00	2.200E-05	3.490E+01
TE-129M	1.090E+05	3.050E+04	1.700E+04	3.520E+04	3.210E+05	1.680E+00	1.330E+05
TE-129	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.140E-06
TE-131M	1.480E+03	5.130E+02	5.460E+02	1.050E+03	4.960E+03	7.280E-01	2.080E+04
TE-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-132	8.150E+03	3.610E+03	4.360E+03	5.260E+03	3.350E+04	4.300E-01	3.630E+04
I-130	6.360E+00	1.270E+01	6.620E+00	1.390E+03	1.900E+01	1.130E-01	6.020E+00
I-131	2.060E+03	2.070E+03	1.180E+03	6.840E+05	3.400E+03	1.510E+00	1.860E+02
I-132	3.600E-04	4.780E-04	2.200E-04	2.220E-02	7.320E-04	0.000E+00	5.630E-04
I-133	4.650E+01	5.740E+01	2.180E+01	1.060E+04	9.560E+01	1.490E-01	2.320E+01
I-134	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-135	3.790E-01	6.820E-01	3.230E-01	6.040E+01	1.050E+00	6.240E-05	5.200E-01
CS-134	7.610E+05	1.250E+06	2.640E+05	2.850E+02	3.870E+05	1.390E+05	7.020E+03
CS-136	4.140E+04	1.140E+05	7.370E+04	3.090E+01	6.060E+04	9.050E+03	4.010E+03
CS-137	1.120E+06	1.070E+06	1.590E+05	4.250E+02	3.500E+05	1.260E+05	7.150E+03
CS-138	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.510E-05
BA-140	8.640E+03	8.950E+00	5.050E+02	1.380E+00	3.850E+00	5.890E+00	4.380E+03
BA-141	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	8.300E-01	4.590E-01	3.270E-01	2.590E-01	2.590E-01	2.590E-01	5.560E+03
LA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.990E+04
CE-141	2.040E+01	1.050E+01	2.180E+00	7.210E-01	5.030E+00	7.210E-01	1.230E+04
CE-143	4.850E-01	1.420E+02	2.440E-01	2.230E-01	2.830E-01	2.230E-01	2.080E+03
CE-144	2.390E+03	7.510E+02	1.300E+02	2.970E+00	4.170E+02	2.970E+00	1.950E+05
PR-143	1.110E+01	3.330E+00	5.510E-01	0.000E+00	1.800E+00	0.000E+00	1.200E+04
PR-144	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND-147	7.380E+00	6.040E+00	7.880E-01	3.470E-01	3.470E+00	3.470E-01	9.020E+03
W-187	2.650E+01	1.580E+01	7.170E+00	1.770E-01	1.770E-01	1.770E-01	2.190E+03
NP-239	4.000E-01	2.030E-01	1.990E-01	1.880E-01	2.320E-01	1.880E-01	1.130E+03

ORGAN DOSE FACTORS FOR LIQUID EFFLUENTS (Continued)

DOSE FACTOR TABLE: A(') - Infant

Units are mrem/hr per uCi/ml

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	1.140E+01	1.140E+01	[illegible]40E+01	1.140E+01	1.140E+01	1.140E+01
C-14	5.800E+05	1.240E+05	1.240E+05	[illegible]40E+05	1.240E+05	1.240E+05	1.240E+05
NA-24	1.760E+02	1.760E+02	1.760E+02	1.760E+02	1.760E+02	1.760E+02	1.760E+02
P-32	9.000E+05	5.290E+04	3.490E+04	0.000E+00	0.000E+00	0.000E+00	1.220E+04
CR-51	0.000E+00	0.000E+00	7.820E-01	5.100E-01	1.120E-01	9.930E-01	2.280E+01
MN-54	0.000E+00	1.680E+02	3.820E+01	0.000E+00	3.730E+01	0.000E+00	6.180E+01
MN-56	0.000E+00	1.220E-06	0.000E+00	0.000E+00	1.050E-06	0.000E+00	1.110E-04
FE-55	5.610E+02	3.630E+02	9.690E+01	0.000E+00	0.000E+00	1.770E+02	4.610E+01
FE-59	1.040E+03	1.810E+03	7.140E+02	0.000E+00	0.000E+00	5.360E+02	8.650E+02
CO-58	0.000E+00	1.080E+02	2.700E+02	0.000E+00	0.000E+00	0.000E+00	2.700E+02
CO-60	0.000E+00	3.810E+02	8.990E+02	0.000E+00	0.000E+00	0.000E+00	9.060E+02
NI-63	1.720E+05	1.060E+04	5.970E+03	0.000E+00	0.000E+00	0.000E+00	5.290E+02
NI-65	1.470E-04	1.660E-05	7.560E-06	0.000E+00	0.000E+00	0.000E+00	1.270E-03
CU-64	0.000E+00	2.290E+00	1.060E+00	0.000E+00	3.870E+00	0.000E+00	4.700E+01
ZN-65	3.020E+04	1.030E+05	4.770E+04	0.000E+00	5.020E+04	0.000E+00	8.740E+04
ZN-69	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-83	0.000E+00	0.000E+00	4.150E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-84	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BR-85	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-86	0.000E+00	1.140E+05	5.650E+04	0.000E+00	0.000E+00	0.000E+00	2.920E+03
RB-88	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RB-89	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
SR-89	5.770E+04	0.000E+00	1.660E+03	0.000E+00	0.000E+00	0.000E+00	1.190E+03
SR-90	3.910E+05	0.000E+00	1.080E+05	0.000E+00	0.000E+00	0.000E+00	7.230E+03
SR-91	4.010E+00	0.000E+00	1.450E-01	0.000E+00	0.000E+00	0.000E+00	4.750E+00
SR-92	[illegible]10E-04	0.000E+00	6.830E-06	0.000E+00	0.000E+00	0.000E+00	1.980E-03
Y-90	4.550E-03	0.000E+00	1.220E-04	0.000E+00	0.000E+00	0.000E+00	6.280E+00
Y-91M	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Y-91	3.320E-01	0.000E+00	8.830E-03	0.000E+00	0.000E+00	0.000E+00	2.380E+01
Y-92	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	3.310E-04
Y-93	3.060E-05	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	2.420E-01
ZR-95	3.060E-02	7.450E-03	5.280E-03	0.000E+00	8.030E-03	0.000E+00	3.710E+00
ZR-97	4.310E-05	7.400E-06	3.380E-06	0.000E+00	7.460E-06	0.000E+00	4.720E-01
NB-95	2.610E+00	1.160E+00	6.700E-01	0.000E+00	8.310E-01	0.000E+00	9.780E+02
MO-99	0.000E+00	1.380E+03	2.690E+02	0.000E+00	2.060E+03	0.000E+00	4.550E+02
TC-99M	5.610E-04	1.160E-03	1.490E-02	0.000E+00	1.250E-02	6.050E-04	3.360E-01
TC-101	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
RU-103	4.080E-02	0.000E+00	1.370E-02	0.000E+00	8.500E-02	0.000E+00	4.970E-01
RU-105	0.000E+00	0.000E+00	0.000E+00	0.000E+00	1.520E-06	0.000E+00	8.250E-05
RU-106	8.400E-01	0.000E+00	1.050E-01	0.000E+00	9.940E-01	0.000E+00	6.380E+00
AG-110M	1.810E+03	1.320E+03	8.740E+02	0.000E+00	1.890E+03	0.000E+00	6.850E+04
TE-125M	8.480E+03	2.830E+02	1.150E+02	2.850E+02	0.000E+00	0.000E+00	4.040E+02
TE-127M	2.620E+03	8.690E+02	3.170E+02	7.570E+02	6.450E+03	0.000E+00	1.060E+03
TE-127	9.610E-02	3.220E-02	2.070E-02	7.820E-02	2.340E-01	0.000E+00	2.020E+00
TE-129M	3.050E+03	1.050E+03	4.700E+02	1.170E+03	7.630E+03	0.000E+00	1.820E+03
TE-129	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-131M	2.870E+01	1.160E+01	9.550E+00	2.340E+01	7.960E+01	0.000E+00	1.950E+02
TE-131	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
TE-132	1.410E+02	6.980E+01	6.510E+01	1.030E+02	4.360E+02	0.000E+00	2.580E+02
I-130	8.920E+00	1.960E+01	7.880E+00	2.200E+03	2.160E+01	0.000E+00	4.210E+00
I-131	3.070E+03	3.610E+03	1.590E+03	1.190E+06	4.220E+03	0.000E+00	1.290E+02
I-132	1.300E-05	2.640E-05	9.410E-06	1.240E-03	2.950E-05	0.000E+00	2.140E-05
I-133	6.990E+01	1.020E+02	2.980E+01	1.850E+04	1.200E+02	0.000E+00	1.720E+01
I-134	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
I-135	4.230E-01	8.410E-01	3.070E-01	7.540E+01	9.380E-01	0.000E+00	3.040E-01
CS-134	1.550E+05	2.890E+05	2.920E+04	0.000E+00	7.430E+04	3.050E+04	7.840E+02
CS-136	1.050E+04	3.080E+04	1.150E+04	0.000E+00	1.230E+04	2.510E+03	4.670E+02
CS-137	2.320E+05	2.710E+05	1.920E+04	0.000E+00	7.270E+04	2.940E+04	8.470E+02
CS-138	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-139	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-140	1.280E+03	1.280E+00	6.610E+01	0.000E+00	3.050E-01	7.880E-01	3.150E+02
BA-141	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
BA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
LA-140	3.050E-04	1.200E-04	3.090E-05	0.000E+00	0.000E+00	0.000E+00	1.410E+00
LA-142	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
CE-141	2.070E-01	1.260E-01	1.490E-02	0.000E+00	3.890E-02	0.000E+00	6.520E+01
CE-143	3.200E-03	2.120E+00	2.420E-04	0.000E+00	6.180E-04	0.000E+00	1.240E+01
CE-144	9.820E+00	4.020E+00	5.500E-01	0.000E+00	1.620E+00	0.000E+00	5.630E+02
PR-143	7.860E-03	2.940E-03	3.900E-04	0.000E+00	1.090E-03	0.000E+00	4.150E+00
PR-144	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
ND-147	4.810E-03	4.940E-03	3.020E-04	0.000E+00	1.900E-03	0.000E+00	3.130E+00
W-187	5.550E-01	3.860E-01	1.330E-01	0.000E+00	0.000E+00	0.000E+00	2.270E+01
NP-239	2.510E-04	2.240E-05	1.270E-05	0.000E+00	4.4[illegible]E-05	0.000E+00	6.480E-01

HISTORICAL GASEOUS SOURCE TERMS

	Continuous Mode*			Batch Mode**		
		Relative			Relative	
Nuclide	C_i (µCi/cc)	%	fraction	C_i (µCi/cc)	%	fraction
Kr-85	0	0	0	2.89E-08	.23	0.0023
Kr-85M	6.89E-09	.36	0.0036	0	0	0
Kr-88	1.37E-08	.71	0.0071	0	0	0
Xe-131M	0	0	0	1.04E-07	.81	0.0081
Xe-133	1.77E-06	92.03	0.9203	1.26E-05	98.35	0.9835
Xe-133M	2.47E-08	1.28	0.0128	7.70E-08	.60	0.0060
Xe-135	1.08E-07	5.62	0.0562	1.70E-09	.01	0.0001
TOTAL	1.92E-06	100.00	1.0	1.28E-05	100.00	1.0

Based on effluent history of Rancho Seco (January-June 1989 SRERR)

* This source term is applicable for ABS and ABGLV.

** This source term is applicable for RBS (initial and continuing releases) and WGDT.

GASEOUS MONITOR DETECTOR EFFICIENCIES

R15044, R15045, and R15546A
(Detector Model RD-52)

Nuclide	Efficiency cpm/μCi/cc
KR-83M	0.0
KR-85M	7.30E+07
KR-85	7.19E+07
KR-87	8.70E+07
KR-88	8.70E+07
KR-89	8.70E+07
KR-90	8.70E+07
XE-131M	0.0
XE-133M	0.0
XE-133	2.94E+07
XE-135M	0.0
XE-135	7.75E+07
XE-137	8.70E+07
XE-138	8.70E+07
AR-41	7.80E+07

* From Calculation Z-RDM-I0261

DOSE FACTORS FOR NOBLE GASES AND DAUGHTERS

FACTORS FOR EXPOSURE TO A SEMI-INFINITE CLOUD
OF NOBLE GASES

	Dose to People *		Dose to Air #	
Nuclide	Gamma-Body K(i)	Beta-Skin L(i)	Gamma M(i)	Beta N(i)
AR-41	8.840E+03	2.690E+03	9.300E+03	3.280E+03
KR-83M	7.560E-02	0.000E+00	1.930E+01	2.880E+02
KR-85	1.610E+01	1.340E+03	1.720E+01	1.950E+03
KR-85M	1.170E+03	1.460E+03	1.230E+03	1.970E+03
KR-87	5.920E+03	9.730E+03	6.170E+03	1.030E+04
KR-88	1.470E+04	2.370E+03	1.520E+04	2.930E+03
KR-89	1.660E+04	1.010E+04	1.730E+04	1.060E+04
KR-90	1.560E+04	7.290E+03	1.630E+04	7.830E+03
XE-131M	9.150E+01	4.760E+02	1.560E+02	1.110E+03
XE-133	2.940E+02	3.060E+02	3.530E+02	1.050E+03
XE-133M	2.510E+02	9.940E+02	3.270E+02	1.480E+03
XE-135	1.810E+03	1.860E+03	1.920E+03	2.460E+03
XE-135M	3.120E+03	7.110E+02	3.360E+03	7.390E+02
XE-137	1.420E+03	1.220E+04	1.510E+03	1.270E+04
XE-138	8.830E+03	4.130E+03	9.210E+03	4.750E+03

* -- mrem/yr per uCi/cu.m

-- mrad/yr per uCi/cu.m

ORGAN DOSE FACTORS FOR GASEOUS EFFLUENTS

DOSE FACTOR TABLE: R(I) - Adult, inhalation

Units are mrem/yr per uCi/cu.m

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	1.260E+03	1.260E+03	1.260E+03	1.260E+03	1.260E+03	1.260E+03
CR-51	0.000E+00	0.000E+00	1.000E+02	5.950E+01	2.280E+01	1.440E+04	3.320E+03
MN-54	0.000E+00	3.960E+04	6.300E+03	0.000E+00	9.840E+03	1.400E+06	7.740E+04
FE-55	2.460E+04	1.700E+04	3.950E+03	0.000E+00	0.000E+00	7.210E+04	6.030E+03
FE-59	1.180E+04	2.780E+04	1.060E+04	0.000E+00	0.000E+00	1.020E+06	1.880E+05
CO-58	0.000E+00	1.580E+03	2.070E+03	0.000E+00	0.000E+00	9.280E+05	1.060E+05
CO-60	0.000E+00	1.150E+04	1.480E+04	0.000E+00	0.000E+00	5.970E+06	2.850E+05
ZN-65	3.240E+04	1.030E+05	4.660E+04	0.000E+00	6.900E+04	8.640E+05	5.350E+04
SR-89	3.040E+05	0.000E+00	8.720E+03	0.000E+00	0.000E+00	1.400E+06	3.500E+05
SR-90	9.920E+07	0.000E+00	6.100E+06	0.000E+00	0.000E+00	9.600E+06	7.220E+05
ZR-95	1.070E+05	3.440E+04	2.330E+04	0.000E+00	5.420E+04	1.770E+06	1.500E+05
I-131	2.520E+04	3.580E+04	2.050E+04	1.190E+07	6.130E+04	0.000E+00	6.280E+03
I-133	8.640E+03	1.480E+04	4.520E+03	2.150E+06	2.590E+04	0.000E+00	8.880E+03
CS-134	3.730E+05	8.480E+05	7.280E+05	0.000E+00	2.870E+05	9.760E+04	1.040E+04
CS-136	3.910E+04	1.460E+05	1.100E+05	0.000E+00	8.560E+04	1.200E+04	1.170E+04
CS-137	4.790E+05	6.210E+05	4.280E+05	0.000E+00	2.230E+05	7.520E+04	8.400E+03
BA-140	3.910E+04	4.910E+01	2.570E+03	0.000E+00	1.670E+01	1.270E+06	2.180E+05
CE-141	1.990E+04	1.350E+04	1.530E+03	0.000E+00	6.270E+03	3.620E+05	1.200E+05
CE-144	3.430E+06	1.430E+06	1.840E+05	0.000E+00	8.480E+05	7.780E+06	8.160E+05

DOSE FACTOR TABLE: R(I) - Teen, inhalation,

Units are mrem/yr per uCi/cu.m

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	1.270E+03	1.270E+03	1.270E+03	1.270E+03	1.270E+03	1.270E+03
CR-51	0.000E+00	0.000E+00	1.350E+02	7.500E+01	3.070E+01	2.100E+04	3.000E+03
MN-54	0.000E+00	5.110E+04	8.400E+03	0.000E+00	1.270E+04	1.980E+06	6.680E+04
FE-55	3.350E+04	2.390E+04	5.550E+03	0.000E+00	0.000E+00	1.240E+05	6.390E+03
FE-59	1.590E+04	3.700E+04	1.430E+04	0.000E+00	0.000E+00	1.530E+06	1.780E+05
CO-58	0.000E+00	2.070E+03	2.780E+03	0.000E+00	0.000E+00	1.340E+06	9.520E+04
CO-60	0.000E+00	1.510E+04	1.980E+04	0.000E+00	0.000E+00	8.720E+06	2.590E+05
ZN-65	3.860E+04	1.340E+05	6.240E+04	0.000E+00	8.640E+04	1.240E+06	4.670E+04
SR-89	4.350E+05	0.000E+00	1.250E+04	0.000E+00	0.000E+00	2.420E+06	3.710E+05
SR-90	1.080E+08	0.000E+00	6.680E+06	0.000E+00	0.000E+00	1.650E+07	7.650E+05
ZR-95	1.460E+05	4.590E+04	3.150E+04	0.000E+00	6.740E+04	2.690E+06	1.490E+05
I-131	3.550E+04	4.910E+04	2.640E+04	1.460E+07	8.400E+04	0.000E+00	6.490E+03
I-133	1.220E+04	2.050E+04	6.230E+03	2.920E+06	3.590E+04	0.000E+00	1.030E+04
CS-134	5.030E+05	1.130E+06	5.490E+05	0.000E+00	3.750E+05	1.460E+05	9.760E+03
CS-136	5.150E+04	1.940E+05	1.370E+05	0.000E+00	1.100E+05	1.780E+04	1.090E+04
CS-137	6.710E+05	8.480E+05	3.110E+05	0.000E+00	3.040E+05	1.210E+05	8.480E+03
BA-140	5.470E+04	6.710E+01	3.520E+03	0.000E+00	2.280E+01	2.030E+06	2.290E+05
CE-141	2.840E+04	1.900E+04	2.170E+03	0.000E+00	8.880E+03	6.140E+05	1.260E+05
CE-144	4.890E+06	2.020E+06	2.630E+05	0.000E+00	1.210E+06	1.340E+07	8.640E+05

ORGAN DOSE FACTORS FOR GASEOUS EFFLUENTS

DOSE FACTOR TABLE: R(i) - Child, inhalation

Units are mrem/yr per uCi/cu.m

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	1.130E+03	1.130E+03	1.130E+03	1.130E+03	1.130E+03	1.130E+03
CR-51	0.000E+00	0.000E+00	1.540E+02	8.550E+01	2.430E+01	1.700E+04	1.080E+03
MN-54	0.000E+00	4.29[illegible]+04	9.510E+03	0.000E+00	1.000E+04	1.580E+06	2.290E+04
FE-55	4.740E+04	2.520E+04	7.770E+03	0.000E+00	0.000E+00	1.110E+05	2.870E+03
FE-59	2.070E+04	3.350E+04	1.670E+04	0.000E+00	0.000E+00	1.270E+06	7.070E+04
CO-58	0.000E+00	1.770E+03	3.160E+03	0.000E+00	0.000E+00	1.110E+06	3.440E+04
CO-60	0.000E+00	1.310E+04	2.270E+04	0.000E+00	0.000E+00	7.070E+06	9.620E+04
ZN-65	4.260E+04	1.130E+05	7.030E+04	0.000E+00	7.140E+04	9.960E+05	1.630E+04
SR-89	6.000E+05	0.000E+00	1.720E+04	0.000E+00	0.000E+00	2.160E+06	1.670E+05
SR-90	1.010E+08	0.000E+00	6.440E+06	0.000E+00	0.000E+00	1.480E+07	3.440E+05
ZR-95	1.900E+05	4.180E+04	3.700E+04	0.000E+00	5.960E+04	2.230E+06	6.110E+04
I-131	4.810E+04	4.810E+04	2.730E+04	1.630E+07	7.880E+04	0.000E+00	2.840E+03
I-133	1.660E+04	2.030E+04	7.700E+03	3.850E+06	3.380E+04	0.000E+00	5.480E+03
CS-134	6.510E+05	1.010E+06	2.250E+05	0.000E+00	3.310E+05	1.210E+05	3.850E+03
CS-136	6.510E+04	1.710E+05	1.160E+05	0.000E+00	9.550E+04	1.450E+04	4.180E+03
CS-137	9.070E+05	8.250E+05	1.28[illegible]+05	0.000E+00	2.820E+05	1.040E+05	3.620E+03
BA-140	7.40[illegible]04	6.480E+01	4.330E+03	0.000E+00	2.110E+01	1.740E+06	1.020E+05
CE-141	3.920E+04	1.950E+04	[illegible].900E+03	0.000E+00	8.550E+03	5.440E+05	5.660E+04
CE-144	6.770E+06	2.120E+06	3.620E+05	0.000E+00	1.170E+06	1.200E+07	3.890E+05

DOSE FACTOR TABLE: R(i) - Infant, inhalation

Units are mrem/yr per uCi/cu.m

Nuclide	Bone	Liver	Tbody	Thyroid	Kidney	Lung	Gitract
H-3	0.000E+00	6.470E+02	6.470E+02	6.470E+02	6.470E+02	6.470E+02	6.470E+02
CR-51	0.000E+00	0.000E+00	8.950E+01	5.760E+01	1.320E+01	1.280E+04	3.570E+02
MN-54	0.000E+00	2.540E+04	4.990E+03	0.000E+00	4.990E+03	1.000E+06	7.060E+03
FE-55	1.970E+04	1.180E+04	3.330E+03	0.000E+00	0.000E+00	8.700E+04	1.100E+03
FE-59	1.360E+04	2.350E+04	9.480E+03	0.000E+00	0.000E+00	1.020E+06	2.480E+04
CO-58	0.000E+00	1.220E+03	1.820E+03	0.000E+00	0.000E+00	7.770E+05	1.110E+04
CO-60	0.000E+00	8.030E+03	1.180E+04	0.000E+00	0.000E+00	4.510E+06	3.190E+04
ZN-65	1.930E+04	6.260E+04	3.110E+04	0.000E+00	3.250E+04	6.470E+05	5.140E+04
SR-89	3.980E+05	0.000E+00	1.140E+04	0.000E+00	0.000E+00	2.030E+06	6.400E+04
SR-90	4.090E+07	0.000E+00	2.590E+06	0.000E+00	0.000E+00	1.120E+07	1.310E+05
ZR-95	1.150E+05	2.790E+04	2.030E+04	0.000E+00	3.110E+04	1.750E+06	2.170E+04
I-131	3.800E+04	4.440E+04	1.960E+04	1.480E+07	5.180E+04	0.000E+00	1.060E+03
I-133	1.320E+04	1.920E+04	5.600E+03	3.560E+06	2.240E+04	0.000E+00	2.160E+03
CS-134	3.960E+05	7.030E+05	7.450E+04	0.000E+00	1.900E+05	7.970E+04	1.330E+03
CS-136	4.830E+04	1.350E+05	5.290E+04	0.000E+00	5.640E+04	1.180E+04	1.430E+03
CS-137	5.490E+05	6.120E+05	4.550E+04	0.000E+00	1.720E+05	7.130E+04	1.330E+03
BA-140	5.600E+04	5.600E+01	2.900E+03	0.000E+00	1.340E+01	1.600E+06	3.840E+04
CE-141	2.770E+04	1.670E+04	1.990E+03	0.000E+00	5.250E+03	5.170E+05	2.160E+04
CE-144	3.190E+06	1.210E+06	1.760E+05	0.000E+00	5.380E+05	9.850E+06	1.480E+05

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
A) Retention Basin Effluent Discharge Monitor	1	With the monitor inoperable, effluent releases may be resumed provided that prior to initiating a release from the retention basin: 1) At least two independent samples are analyzed in accordance with Step 6.14.2. 2) At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via the pathway. Exert best efforts to return the monitor to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semi-Annual Radioactive Effluent Release Report why the inoperable monitor was not restored in a timely manner.

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

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
1) Gross Radioactivity Monitors Providing Alarm and Automatic Isolation				
A) Retention Basin Effluent Discharge Monitor	$D^{(1)}$	P	$R^{(2)}$	$Q^{(3)}$
2) Flow Monitors				
A) Regenerant Holdup Tank Discharge Line Total Flow	$D^{(4)}$	N/A	R	Q
B) Waste Water Flow Rate and Totalizer	$D^{(4)}$	N/A	R	Q

Table Notation

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

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

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

A. Instrumentation indicates measured levels above the alarm/trip setpoint.

B. Circuit failure.

C. Instrument indicates a downscale failure.

D. Instrument controls not set in operate mode.

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

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (a) (uCi/ml)
A. Retention Basin N/S(b)	Each Batch P	Each Batch P	Mn-54, Fe-59 Co-58, Co-60 Zn-65, Mo-99 Cs-134, Cs-136 Cs-137, Ba-140 Cs-141, Ce-144 I-131	3E-7
			Dissolved and Entrained Noble Gases (Gamma Emitters) H-3	1E-5
B. Regenerant Holdup Tank A/B(c,d)	Each Batch	Each Batch	Mn-54, Fe-59 Co-58, Co-60 Zn-65, Cs-134 Cs-137, Ce-141 I-131	2E-8
			Cs-136	8E-9
			Mo-99, Ba-140 Ce-144	6E-8
			H-3	1E-5
	Each Batch P	Composite(e) M	Cs-134, Cs-137	3E-9
			Mn-54, Co-58 Co-60	4E-9
			Zn-65	6E-9
			Fe-59	8E-9
			Sr-89	5E-9
			Sr-90	1E-9
			Gross Alpha	1E-7

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
(Continued)

Table Notation

(a) 1. The Lower Limits of Detection (LLDs) for the radionuclides presented in this table are the smallest concentrations (expressed in microcuries per milliliter) which are required to be detected, if present, in order to achieve compliance with the limits of Step 6.14.3 (10 CFR 50, Appendix I) for a RHUT transfer to a retention basin and assurance of compliance with the limits of Step 6.14.2 (10 CFR 20, Appendix B, Table II, Column 2) for a Retention Basin Discharge.

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

$$LLD = \frac{2.71/t_s + 3.29S_b}{3.70E4(YEV)\exp(-\lambda t_c)}$$

Where:

2.71 = factor to account for Poisson statistics at very low background count rates, and 3.29 = two times the constant used to establish the one sided 0.95 confidence interval.

S_b = the standard deviation of the background counting rate

$= (B/(t_b t_s) + B/t_b^2)^{0.5}$

Where:

B = background counts

t_b = background counting interval (seconds)

t_s = sample counting interval (seconds)

3.70E4 = disintegrations/second/microcurie

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

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM
(Continued)

Table Notation (Continued)

E = counting efficiency (count/disintegrations)

V = sample volume (milliliters)

λ = the radioactive decay constant for the particular nuclide (seconds^{-1})

t_c = the elapsed time from midpoint of collection to the midpoint of counting

3. The LLD is defined as an **a priori** (before the fact) estimate and is not to be calculated for each sample analyzed on an **a** posteriori (after the fact) basis.

(b) A batch release is the discharge of liquid wastes of discrete volume from the north or south Retention Basin. The Retention Basins are the maximum permissible concentration accountability points for 10 CFR 20, Appendix B compliance.

(c) A RHUT will be isolated and its contents thoroughly mixed to assure representative sampling prior to transferring the contents to a Retention Basin. The A and B RHUTs are the dose equivalent accountability points for 10 CFR 50, Appendix I compliance. The RHUT pre-release Low Limits of Detection equate to an offsite dose of less than 50 percent of the 10 CFR 50, Appendix I guidelines. The monthly RHUT composite Lower Limits of Detection equate to an offsite dose of less than 10 percent of the 10 CFR 50, Appendix I guidelines. These Lower Limits of Detection may change as the maximum annual effluent outflow or the minimum annual average flow rate in the plant effluent stream changes.

(d) Isotopic peaks which are measurable and identifiable from a RHUT pre-release sample analysis shall be reported and included in ODCM evaluations. Nuclides which are not observed in the analysis shall be reported as "less than" the nuclide's **a posteriori** minimum detectable concentration and shall not be reported as being present. The "less than" results shall be considered "zero" for the purposes of ODCM evaluations; however, if a nuclide is measured and identified at a value less than the Attachment 16 LLD value, it shall be reported and entered in ODCM evaluations.

(e) A composite sample shall be obtained by mixing liquid aliquot volumes in proportion to the volume of liquid released from each RHUT.

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent		
a. Noble Gas Activity Monitor providing alarm and automatic termination of release.	1	With the monitor channel alarm/trip setpoint less conservative than the setpoint calculated as described in Step 6.6, immediately suspend the release or declare the channel inoperable. With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed in accordance with Attachment 19 within 24 hours.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Attachment 19 within 24 hours.*
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Attachment 19 within 24 hours.*

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
(Continued)

Instrument	Minimum Number of Channels Operable	Action
1. Reactor Building Purge Vent (Continued)		
d. System Effluent Flow Device	1	With the flow rate device inoperable, effluent releases may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measurement Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.
2. Auxiliary Building Stack		
a. Noble Gas Activity Monitor providing alarm and automatic termination of Waste Gas Header release.	1	With the monitor alarm/trip setpoint less conservative than the setpoint calculated as described in Step 6.6, immediately suspend the release or declare the channel inoperable. With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed in accordance with Attachment 19 within 24 hours. With the monitor inoperable, the contents of the Waste Gas System tank(s) may be released to the environment provided that prior to initiating the release: a. At least two independent samples of the tank's contents are analyzed, and

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
(Continued)

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack (Continued)		
		b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup; Otherwise, suspend release of radioactive effluents via this pathway.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Attachment 19 within 24 hours.*
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Attachment 19 within 24 hours.*
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate used is the maximum design flow rate.

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
(Continued)

Instrument	Minimum Number of Channels Operable	Action
2. Auxiliary Building Stack (Continued)		
e. Sampler Flow Rate Measuring Devices	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.
3. Auxiliary Building Grade Level Vent		
a. Noble Gas Activity Monitor	1	With the monitor channel alarm/trip setpoint less conservative than the setpoint calculated as described in Step 5.6, immediately suspend the release or declare the channel inoperable. With the monitor inoperable, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed in accordance with Attachment 19 within 24 hours.
b. Iodine Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Attachment 19 within 24 hours.*

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
(Continued)

Instrument	Minimum Number of Channels Operable	Action
3. Auxiliary Building Grade Level Vent (Continued)		
c. Particulate Sampler	1	With the collection device inoperable, effluent releases via this pathway may continue provided continuous samples are taken within one hour after the monitor is declared inoperable and these samples are analyzed in accordance with Attachment 19 within 24 hours.*
d. System Effluent Flow Rate Device	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate used is the maximum design flow rate.
e. Sampler Flow Rate Measuring Devices	1	With the flow rate device inoperable, effluent releases via this pathway may continue provided the flow rate is estimated and recorded at least once per 4 hours.

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS (Continued)

Instrument	Instrument Channel Check	Source Check	Instrument Channel Calibration	Channel Test
3. Auxiliary Building Grade Level Vent (Continued)				
c. Particulate Sampler	W	NA	NA	NA
d. System Effluent Flow Rate Device	D	NA	R	Q
e. Sampler Monitor Flow Rate Measurement Device	D	NA	R	Q

Table Notation

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

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

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

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS (Continued)

Table Notation (Continued)

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

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

(5) Deleted.

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

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) (µCi/ml)
A. Reactor Building Purge and Equalization Vent	P Each Purge and Equalization Vent Grab Sample(b,e,i)	P Each Purge and Equalization Vent(b,e,i)	Principal Gamma Emitters (f)	1E-4
			H-3	1E-6
B. Auxiliary Building Stack	M (b,c,e) Grab Sample	M (b)	Principal Gamma Emitters (f)	1E-4
			H-3	1E-6
C. Auxiliary Building Grade Level Vent	M (b) Grab Sample	M (b)	Principal Gamma Emitters (f)	1E-4
			H-3	1E-6
D. All Release Types as listed in A,B,C above	Continuous	W (d) Charcoal Sample	I-131	1E-12
	Continuous	W (d) Particulate Sample	Principal Gamma Emitters (f) (I-131, Others)	1E-11
	Continuous	M Composite Particulate Sample	Gross Alpha (h)	1E-11
			Sr-89, Sr-90 (g)	1E-11
	Continuous	Noble Gas Monitor	Noble Gases Gross Beta and Gamma	1E-4 as Xe-133

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM
(Continued)

Table Notation

(a) 1. The Lower Limits of Detection (LLDs) for the radionuclides presented in this table are the smallest concentration (expressed in microcuries per unit volume) which are required to be detected, if present, in order to achieve compliance with the limits of the Specifications in Steps 6.14.6, 6.14.7, and 6.14.8.

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

$$LLD = \frac{2.71/t_s + 3.29S_b}{3.70E4(YEV)\exp(-\lambda t_c)}$$

Where 2.71 = factor to account for Poisson statistics at very low background count rates, and 3.29 = two times the constant used to establish the one sided 0.95 confidence interval.

S_b = the standard deviation if the background counting rate

$= (B/(t_b t_s) + B/t_b^2)^{0.5}$

where,

B = background counts

t_b = background counting interval (seconds)

t_s = sample counting interval (seconds)

3.70E4 = disintegrations/second/microcurie

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

E = counting efficiency (counts/disintegration)

V = sample volume (cubic centimeters)

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM
(Continued)

Table Notation (Continued)

λ = the radioactive decay constant for the particular radionuclide (seconds^{-1})

t_c = the elapsed time from midpoint of sample correction to the midpoint of counting (seconds)

3. The LLD is defined as an a priori (before the fact) estimate and is not to be calculated for each sample analyzed on an a posteriori (after the fact) basis.

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

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

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

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

(f) Principal gamma emitters for which the LLD applies are: Kr-87, Mr-88, Xe-133, Xe-133m, Xe-135, Xe-135m, and Xe-138 for gaseous samples and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99 or (or TC-99m), Cs-134, Cs-137, Ce-141, and Ce-144 for particulate samples. This does not mean only these nuclides will be detected and reported. Other peaks that are measurable and identifiable shall be reported in the Semi-Annual Radioactive Effluent Release Report, pursuant to Technical Specification 6.9.2.3. All peaks which are measurable and identifiable shall be reported and entered into the ODCM evaluations. Nuclides which are not observed for the analysis shall be reported as "less

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM
(Continued)

Table Notation (Continued)

than" the nuclide's a posteriori minimum detectable concentration and shall not be reported as being present. The "less than" results shall be considered "zero" for the ODCM evaluations; however, if a nuclide is measured and identified at a value less than the Attachment 19 LLD value, it shall be reported and entered on ODCM evaluations.

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

(h) A gross alpha analysis is performed on a monthly basis for each environmental release particulate sample. This fulfills the requirements of performing a monthly composite.

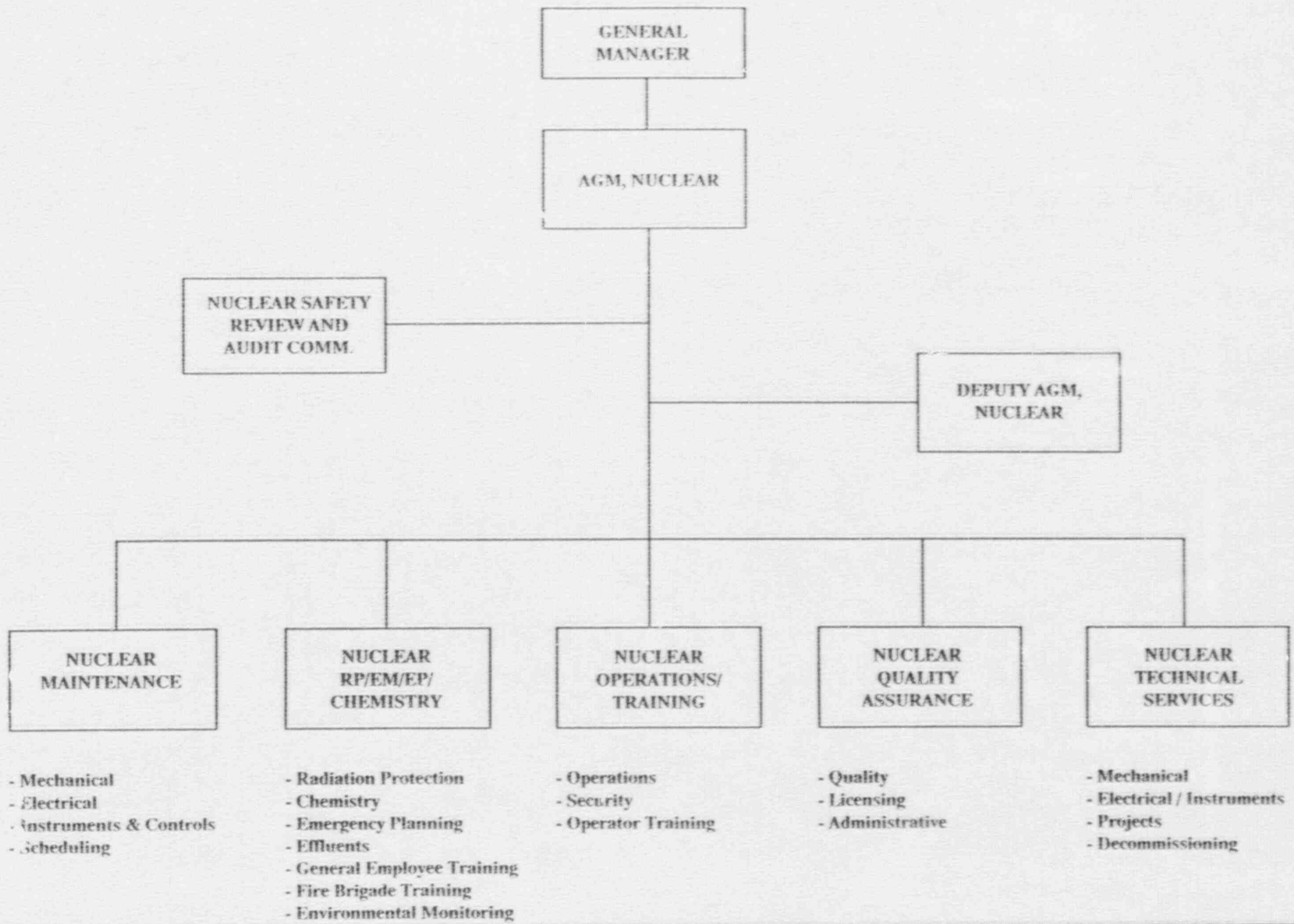
(i) After purging seven reactor building volumes, a technical evaluation prior to initiation of a purge following an out of service period may be conducted in lieu of sampling and analysis.

ATTACHMENT V

1991 NUCLEAR ORGANIZATION

ORG3
R-2

1991 NUCLEAR ORGANIZATION



ATTACHMENT VI

PROCESS CONTROL PROGRAM

RANCHO SECO NUCLEAR GENERATING STATION

PROCESS CONTROL PROGRAM

J.R. Shetler for D. Keuter — 12/27/87

AGM, Nuclear — Date

EFFECTIVE DATE: Initial Issue

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LIST OF FIGURES

DEFINITIONS

Terms in this manual which have a specific definition or meaning are capitalized (e.g., OPERABILITY). Definitions for such capitalized terms are found in Sections D1.0 of Technical Specifications, with the exception of "PROCESSING" which is defined below:

PROCESSING

PROCESSING shall be the conversion of wet radioactive waste into a form that meets shipping and burial ground requirements.

SECTION 1.0

INTRODUCTION

1.1 PURPOSE

The purpose of the Process Control Program (PCP) are to:

(1) Establish a program which will provide reasonable assurance that all radioactive wastes generated at Rancho Seco that are to be disposed of at a land disposal facility are PROCESSED and packaged such that applicable Federal regulations, State rules and regulations, and disposal site criteria are satisfied.

(2) Assure that major changes to the Liquid, Gaseous and Solid Radwaste Systems are properly reviewed, evaluated and approved.

The PCP contains a general description of the methods for controlling the PROCESSING and packaging of solid and liquid radioactive wastes, specific parameters for each method, and the administrative controls and quality assurance required to ensure compliance with applicable regulations and requirements.

1.2 SCOPE

This program defines criteria for the PROCESSING of the following waste streams for disposal at a land disposal facility:

(1) Wet Wastes

- (a) Resins (bead)
- (b) Cartridge Filters
- (c) Evaporator Concentrates
- (d) Sludge
- (e) Miscellaneous liquids

(2) Dry Active Wastes (DAW)

- (a) Compactible
- (b) Noncompactible

This program defines the requirements for major changes to Liquid, Gaseous, and Solid Radwaste Treatment Systems.

1.3 PRECAUTIONS/LIMITATIONS

Except as specifically described in this document, the following general precautions and limitations apply to the PROCESSING and packaging of all radioactive wastes generated at Rancho Seco for disposal at a land disposal facility. These precautions and limitations shall be included in appropriate station or vendor implementing procedures.

(1) No liquid materials within the scope of this program shall be packaged for disposal.

(2) No package shall be loaded for shipment if it has any indication of a hole or failure. These packages shall either be repacked, or placed in an overpack.

(3) Radioactive waste shall not be packaged for disposal in cardboard or fiberboard boxes.

(4) Only High Integrity Containers (HIC) approved for burial at a land disposal facility shall be utilized for packaging dewatered wastes when waste form stability is required per 10 CFR 61.

(5) No objects or materials shall be placed into HICs that may cause chemical or physical damage to the container per the vendors 10 CFR 61 Topical Report or other Federal, State or Burial Facility requirements.

(6) As much as practical, polyethylene HICs shall be kept out of direct sunlight to prevent ultraviolet light degradation. Protection from direct sunlight shall be provided when HICs are stored for extended periods.

(7) Radioactive waste shall not be packaged for disposal if it is pyrophoric. Pyrophoric materials contained in radioactive waste shall be treated, prepared, and packaged to be nonflammable prior to disposal.

(8) Radioactive waste in gaseous form shall not be packaged for disposal.

(9) Radioactive waste containing hazardous material shall be treated to reduce to the maximum extent practicable the potential hazard from the nonradiological materials. Biological, pathogenic or infectious material is not expected to be produced and will be handled on a case by case basis.

(10) Radioactive wastes shall not be packaged for disposal if it is readily capable of detonation or of explosive decomposition or reaction at normal pressures and temperatures, or of explosive reaction with water.

(11) Radioactive waste shall not be packaged for disposal if it contains, or is capable of generating quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling or disposing of the waste.

(12) Samples shall be handled and collected in accordance with applicable Rancho Seco procedures and in keeping with ALARA principles.

(13) PROCESSING evolutions should be periodically monitored for adverse chemical reactions and temperature changes.

1.4 RESPONSIBILITIES

1.4.1 Assistant General Manager, Nuclear

It is the responsibility of the AGM, Nuclear, to ensure that the requirements contained in this manual are achieved during the PROCESSING of radioactive waste by developing appropriate administrative and implementing procedures.

The AGM, Nuclear, is responsible for approving changes to this PROCESS CONTROL PROGRAM, as required by Technical Specification D6.13.

1.4.2 Plant Review Committee

The Plant Review Committee (PRC) shall review the Topical Reports and Process Control Programs of vendors selected to provide waste processing services or products prior to their initial use at the station. Any subsequent revision to these documents shall also be reviewed by PRC prior to the initial use of the revision. These reviews shall ensure compatibility with Station equipment and operation.

Additionally, PRC is responsible for reviewing changes to this PCP prior to implementation. PRC also reviews station administrative and implementing procedures for radioactive waste processing activities.

1.4.3 Radiation Protection Manager

The Radiation Protection Manager (RPM) is responsible for coordinating the review of, and for approving, vendor procedures for waste processing prior to the initial use of the documents at the station. He shall also approve any revisions to these documents prior to initial use of the revision. Review of vendor procedures for waste processing shall include technical review by the Operations, Chemistry, and Nuclear Engineering departments.

SECTION 2.0

RADIOACTIVE WASTE PROCESSING REQUIREMENTS

2.1 SOLID RADIOACTIVE WASTES

2.1.1 Operability Criteria

Radioactive wastes shall be PROCESSED in accordance with this PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

COMPENSATORY MEASURES:

a. With PROCESSING not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct this PROCESS CONTROL PROGRAM, the procedures, and/or the Waste Processing System as necessary to prevent recurrence.

b. With PROCESSING not performed in accordance with this PROCESS CONTROL PROGRAM, test the improperly PROCESSED waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

2.1.2 Tests/Inspections

Satisfactory PROCESSING of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms and boric acid solutions) shall be verified in accordance with this PROCESS CONTROL PROGRAM:

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

This PROCESS CONTROL PROGRAM shall be modified as required, as provided in Technical Specification D6.13, to assure satisfactory PROCESSING of subsequent batches of waste; and

b. With the installed equipment incapable of meeting Operability Criteria 2.1.1, above, or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

2.1.3 <u>Basis</u>

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

2.2 <u>MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS</u>[a]

Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. PRC approved changes shall be retained for review by the Commission. The description for each change shall contain:

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

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

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

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC outside of the RESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) 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 prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the PRC.

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

---

*Licensees may choose to submit the information called for in this Requirement as part of the annual FSAR update.

SECTION 3.0

PROGRAM DESCRIPTION

3.1 PROCESSING OF WET RADIOACTIVE WASTE

3.1.1 PROCESSING Methods

Wet radioactive waste generated at Rancho Seco shall be PROCESSED into a form acceptable for disposal at a licensed facility by dewatering, drying or solidification. PROCESSING shall be performed utilizing vendor supplied services and equipment operating in accordance with vendor's Process Control Program (PCP) and procedures. Any vendor selected to provide services or products used in compliance with 10 CFR 61 stability requirements shall have a topical report addressing 10 CFR 61 requirements under review or approved by the NRC. When a vendor is initially selected, this document shall be revised to incorporate by reference the vendor's Topical Report and PCP. The Topical Reports and PCPs of multiple vendors may be referenced in this PCP even if all vendors are not actively providing services or products at Rancho Seco. However, if any vendor is selected whose documents are not referenced, this document shall be revised to reference them.

3.1.2 Processing System Description

Detailed descriptions of the vendor's PROCESSING system shall be included in the vendor's Topical Report.

3.1.3 Prequalification Testing

Prequalification tests shall be performed on each type of wet radioactive waste stream to demonstrate the ability of the process to produce an acceptable waste form per the requirements of 10 CFR 61. This prequalification testing is performed by the vendor and documented in the vendor's Topical Report.

3.1.4 System Qualification Tests

Prior to the initial PROCESSING of a given waste stream type using a specified process, a test shall be conducted to demonstrate the ability of the process system to produce an acceptable waste form over the range of critical parameters identified during the prequalification testing. Bounds for critical parameters and specific operating limits shall be specified in the vendor's PCP.

These tests shall be performed on laboratory scale or full scale specimens and shall ensure that the acceptance criteria specified in Section 3.1.8 are achieved.

3.1.5 Equipment/System Operability Requirements

Prior to each PROCESSING evolution, the vendor shall demonstrate operability of the processing equipment which shall include but not be limited to the following:

(1) Control Panel
(2) Instrumentation and Controls
(3) Mechanical Equipment
(4) Electrical Equipment

The operability test shall be performed in accordance with station procedures and the vendor's PCP and procedures.

3.1.6 Batch Preprocessing Sampling

Each batch of waste offered for PROCESSING shall be sampled and analyzed, as appropriate, in accordance with station procedures and the vendor's PCP and the Topical Report that addresses the 10 CFR 61 stability requirements. This sampling shall:

(1) Provide necessary data to estimate curie content and perform the waste classification analysis.

(2) Ensure that waste stream parameters are within the bounds for critical parameters established in the vendor's PCP and 10 CFR 61 Topical Report.

NOTE: Results of waste stream chemical analyses shall be reviewed by the PROCESSING vendor to ensure chemical constituents do not exist which could cause adverse chemical reactions during the dewatering process or react adversely with the waste container.

3.1.7 Testing/Inspections

To satisfy the Test/Inspection requirements of Section 2.1.2 of this Program, a test or inspection shall be performed for each PROCESSING evolution to ensure that the applicable acceptance criteria of Section 3.1.8 are achieved. This is accomplished by verifying that a specified end point is achieved for each PROCESSING evolution of the actual waste stream.

If the test results fail to meet the acceptance criteria, the following steps, as per the Compensatory Measures of Section 2.1.1, shall be followed:

(1) PROCESSING of the batch under test shall be suspended until such time as additional testing can be performed, alternative PROCESSING parameters can be determined, and subsequent testing verifies satisfactory PROCESSING of the

waste. PROCESSING of the batch may then be resumed using the alternative PROCESSING parameters if the alternative parameters will produce a product that falls within the vendors qualification envelope.

The vendor's PCP shall provide the method for determining the alternative PROCESSING parameters. Alternative PROCESSING parameters which fall within the vendor's qualification testing envelope shall be approved by the Radwaste Superintendent or his designee and shall be documented in accordance with the vendor's PCP. The vendor's PCP shall be modified as required to assure adequate PROCESSING of subsequent batches of waste. Any changes should be consistent with the conditions, limitations, and restrictions addressed in the vendor's 10 CFR 61 Topical Report.

(2) If the test results failure is due to malfunction of the processing equipment or the processing equipment is inoperable, the equipment shall be returned to an OPERABLE condition or an alternate vendor shall be obtained to process waste as necessary to satisfy applicable transportation and disposal requirements.

3.1.8 Acceptance Criteria

(1) Non-Stable Waste Form

For wastes PROCESSED in non-HICs, the acceptance criteria shall be less than or equal to one-half of one percent (0.5%) of the internal volume of the container for free standing water.

(2) Stable Waste Form

Waste shall be PROCESSED in an approved HIC, and the acceptance criteria shall be less than or equal to one percent (1%) of the internal volume of the HIC for free standing water.

OR

Waste shall be PROCESSED in a non-HIC using a solidification media approved by the NRC, or submitted to the NRC that shows compliance with 10 CFR 61.

Vendor documents for the dewatering process shall include a specified end-point for each dewatering evolution which ensures that these acceptance criteria are achieved.

Vendor documents for solidification shall include a specified test to verify solidification.

3.1.9 Corrective Actions

With processing not meeting the above acceptance criteria or otherwise not meeting disposal site and shipping and transportation requirements, suspend shipment of the

inadequately PROCESSED waste and correct the Process Control Program, the procedures, and/or the waste processing equipment as necessary to prevent a recurrence. Additionally, an evaluation of similar wastes PROCESSED since the last successful surveillance test shall be conducted to determine the extent of the inadequately processed waste. If such wastes have been shipped for disposal, the disposal site operator shall be contacted and the problem addressed.

If PROCESSING is not performed in accordance with this PCP, the improperly PROCESSED waste shall be tested to ensure that it meets burial ground and shipping requirements. Appropriate corrective actions shall be taken to prevent recurrence.

Disposition of inadequately PROCESSED wastes will be handled on a case-by-case basis.

3.2 PROCESSING OF DRY ACTIVE WASTE

Dry Active Waste (DAW) generated at Rancho Seco shall be processed by segregation, sorting, and/or compaction. Processing of DAW is performed to accomplish the following functions:

(1) Package DAW in a fashion acceptable for disposal at a licensed disposal facility.

(2) Remove constituents not acceptable for disposal as DAW.

(3) Minimize volumes of DAW shipped for disposal by:

(a) removing reuseable and uncontaminated items; and
(b) reducing shipped volumes by compaction.

All processing of DAW at Rancho Seco shall be performed in accordance with approved station procedures or vendor procedures that have been reviewed by the Radiation Protection Manager. Vendor equipment, personnel and procedures may be used for DAW processing and packaging.

The segregation of uncontaminated waste from DAW is performed to minimize volumes of DAW shipped for disposal. In order to provide reasonable assurance that radioactive materials are not released as clean waste, the following requirements shall be included in the segregation program, as discussed in Reference 7.5.3:

(1) Surveys, using equipment and techniques for detecting low levels of radioactivity, shall be made of materials that may be contaminated and that are to be disposed of as clean wastes.

(2) Surveys may be conducted on individual items using portable survey instruments, such as pancake GM probes. However, in all cases, final measurements of each package (e.g., bag or box) of aggregated waste to be released as clean waste shall be

performed to ensure that there has not been an accumulation of radioactive material due to the buildup of multiple quantities of contamination which were nondetectable with portable instrumentation. Final measurements shall be performed using sensitive detectors in a low background area, such as scintillation detectors.

3.3 MIXED WASTE

Mixed Waste is defined as waste that contains constituents that satisfy the definition of radioactive waste, subject to the Atomic Energy Act, and contains hazardous waste that either (1) is listed as hazardous waste in 40 CFR 261, Subpart D, or (2) causes the waste to exhibit any of the hazardous waste characteristics identified in 40 CFR 261, Subpart C. Under current federal law, this waste is subject to dual regulation by the NRC and EPA where both agencies have control over the same waste. Due to the complex regulatory issues that must be resolved pertaining to mixed waste, there are currently no authorized disposal sites in the United States which are licensed to receive and dispose of mixed hazardous and radioactive waste.

Since there is currently no avenue for disposal of mixed waste, efforts shall be made to reduce the generation of such waste at Rancho Seco. To accomplish this, station procedures for chemical control and radioactive waste processing shall include the following requirements:

(1) The Station chemical control program shall include a method to identify hazardous constituents of chemicals/chemical products and to evaluate and authorize any usage of these products in areas where mixed waste generation is likely to occur. This evaluation shall consider the substitution of products which are evaluated as non-hazardous per 40 CFR 261.

(2) Radioactive waste processing procedures shall include provisions for segregation and removal of non-radioactive hazardous constituents. Upon removal, such constituents would be handled as hazardous waste as required by the EPA.

(3) Mixed low-level radioactive waste generated at Rancho Seco shall not be shipped for disposal to a low-level radioactive waste disposal facility unless specific approval for such disposal is granted by the appropriate regulatory agencies. Such wastes shall be stored at Rancho Seco until regulatory changes allow for disposal or they are otherwise approved for disposal by appropriate regulatory agencies.

SECTION 4.0

WASTE CLASSIFICATION AND CHARACTERIZATION

4.1 WASTE CLASSIFICATION

Radioactive waste generated at Rancho Seco shall be classified as Class A, B or C in accordance with the requirements of 10 CFR 61, Section 61.55, using one or more of the classification methods given in the USNRC's "Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification (May 1983)". Waste classification shall be performed in accordance with approved station procedures.

The following specific requirements shall be incorporated in the program for sampling and analysis for waste classification:

(1) Annual analyses or an engineering evaluation shall be performed on representative samples of each waste stream or, alternatively, a process stream associated with the generation of the waste for the nuclides listed in Table 1 and Table 2 of 10 CFR 61, Section 61.55.

(2) The results of these analyses shall be used to develop isotopic abundances and scaling factors for difficult to measure nuclides (i.e., beta emitters and transuranics) based on correlations between those nuclides and more easily measured gamma emitters.

(3) Gamma spectroscopy or gross radioactivity measurements shall be made for each container of waste processed for disposal. Calculational methods for determining the total activity in each container shall be developed which use the results of the gamma spectroscopy or gross activity measurements, and the percent isotopic abundances and scaling factors from the annual analyses, or engineering evaluation.

(4) The classification program shall establish criteria and include provisions for increased frequency for the sampling, analysis and evaluation required by paragraph (1), above, if the cobalt 60 to cesium 137 ratio changes by a factor of 10.

(5) Each package of waste shall be clearly labeled as Class A, Class B, or Class C.

4.2 WASTE CHARACTERISTICS

Waste PROCESSED for disposal at Rancho Seco shall meet the applicable characteristics specified in 10 CFR 61, Section 61.56 Waste classified as Class B, Class C, or Class A waste that will not be segregated from Class B and C wastes at the burial facility shall be processed into a stable waste form. This shall be accomplished by placement into a HIC or use of a solidification media which meets the stability requirements of 10 CFR 61.56, per Section 3.1 of this manual.

The vendor's topical report shall include documentation of testing which verifies that the HIC or solidification media meets these stability requirements. Additionally, Rancho Seco shall comply with Federal or State requirements imposed specifically on an approved HIC or solidification media which limits the type and/or radioactive concentration of the waste to be placed in the approved HIC or solidification media.

SECTION 5.0

SPECIFIC WASTE STREAM PROCESSING DESCRIPTIONS

5.1 WET RADIOACTIVE WASTE STREAMS

5.1.1 Resins

Resins will be accumulated from one or more of the following systems:

(1) Makeup and Purification System
(2) Spent Fuel Pool Cooling System
(3) Miscellaneous Liquid Rad Waste System
(4) Coolant Radwaste System
(5) Condensate Polishing Demineralizer System
(6) Other miscellaneous ion exchange medium as generated

Spent Primary resins are collected in the Spent Resin Storage Tank (SRT). These resins are transferred directly to the vendors processing skid from the SRT. If it is necessary to process secondary condensate polishing resins, they are transferred from the polisher to the vendor's processing skid.

Spent resins will be processed for disposal by dewatering or within an approved Solidification Media. The curie content and waste classification of each resin batch shall be estimated prior to sluicing of the spent resin to the vendor dewatering skid. Based on these estimates, the proper liner or HIC and cask for transportation and disposal are selected. The resin is transferred to the liner or HIC where it is dewatered utilizing vendor supplied dewatering services per Section 3.1 of this manual. A representative sample of the resin is collected for final calculations of curie content and waste classification, unless an actual sample was collected prior to the transfer. Containers are sealed, surveyed, and labeled, as appropriate, and stored in a designated storage area until they are shipped for disposal.

A flow chart of a typical resin processing path is shown on Figure 5-1.

5.1.2 Cartridge Filters

Cartridge filters will be accumulated from one or more of the following systems:

(1) Makeup and Purification System
(2) Spent Fuel Pool Cooling System
(3) Miscellaneous Liquid Radwaste System
(4) Coolant Radwaste System
(5) Miscellaneous

Spent filter cartridges are surveyed for dose rate upon removal from the system. The measured dose rate is used to calculate isotopic content using a dose-to-curie conversion factor and scaling factors per Section 4.1 of this manual. Based on the calculated isotopic content, the waste classification and the appropriate process and container for disposal are determined. Normally filters are placed in a liner or HIC. However, filters may be dried and handled as DAW, if conditions allow. Upon completion of the processing, containers are sealed, surveyed, and labeled, as appropriate, and stored in a designated storage area until they are shipped for disposal.

For purposes of waste classification, isotopic concentrations of filters in a liner or HIC should be determined as calculated over the volume of the cartridge filter itself, rather than averaged over the gross volume of the container.

5.1.3 Evaporator Concentrates

Evaporator concentrates result from operation of evaporators for processing of liquid wastes (e.g., floor drains) and boron recovery. The miscellaneous waste evaporator is available to PROCESS liquid waste. The boron recovery evaporator is used to remove boron from reactor grade water and the boric acid concentrates are normally recycled. If PROCESSING of evaporator concentrates is required, the liquid will be solidified or dried in accordance with a vendor approved PROCESS CONTROL PROGRAM reviewed and approved by the PRC.

A flow chart of the liquid processing path is shown on Figure 5-2.

5.1.4 Sludge

Radioactive sludge is accu[illegible] and handled on a case-by-case basis by periodically remov[illegible] sludge from various tanks and sumps throughout the pl[illegible]. Each batch of sludge is sampled for PCP parameters and isotopic content, chemically conditioned, if necessary, and PROCESSED per Section 3.1 of this manual. Containers are sealed, surveyed, and labeled, as appropriate, and stored in a designated storage area until they are shipped for disposal.

5.1.5 Miscellaneous Liquids

Miscellaneous liquids generated in the station will be collected and processed on a case-by-case basis. Such wastes may include decontamination wastes and chemical wastes collected from the Chemistry Labs. Batches of such waste are isolated, sampled for PCP parameters and isotopic content, chemically conditioned, if necessary, and transferred to the vendor's processing skid where it is PROCESSED per Section 3.1 of this manual. Containers are sealed, surveyed, and labeled, as appropriate, and stored in a designated storage area until they are shipped for disposal.

5.2 DRY ACTIVE WASTE

Dry Active Waste (DAW) consists of radioactively contaminated or activated waste which contains no liquids. DAW may be compactible, such as paper, plastic and protective clothing, or non-compactible, such as tools or plant equipment. This waste is segregated by station workers at the point of generation into receptacles designated for "clean" or "contaminated" trash. "Clean" receptacles are used to collect trash that is potentially not contaminated. "Contaminated" containers are used to collect waste that is known or suspected to be contaminated. However, for purposes of DAW processing, all waste collected in Radiation Controlled Areas (i.e., that collected in both the "clean" and "contaminated" receptacles) is assumed to be contaminated until it is surveyed and proven clean. Bags are collected from the receptacles, surveyed for external dose rate, and taken to a designated sorting area for processing. Bags below a specified dose rate level, per station procedures, may be opened and the contents surveyed individually for radioactivity. Items found to be not contaminated per station procedures, reusable items, and items not acceptable for disposal as DAW are removed. In general, the contents of bags above the specified dose rate level are not surveyed for contamination, but are examined for reusable items and items not acceptable for disposal as DAW. Contaminated items are then disposed of as DAW. Compactible items are collected and compressed into approved strong, tight containers. Noncompactible items are placed directly into approved strong, tight containers. Containers are sealed, surveyed, and labeled, as appropriate and stored in a designated storage area until they are shipped for disposal.

A flow chart of the DAW processing path is shown on Figure 5-3.

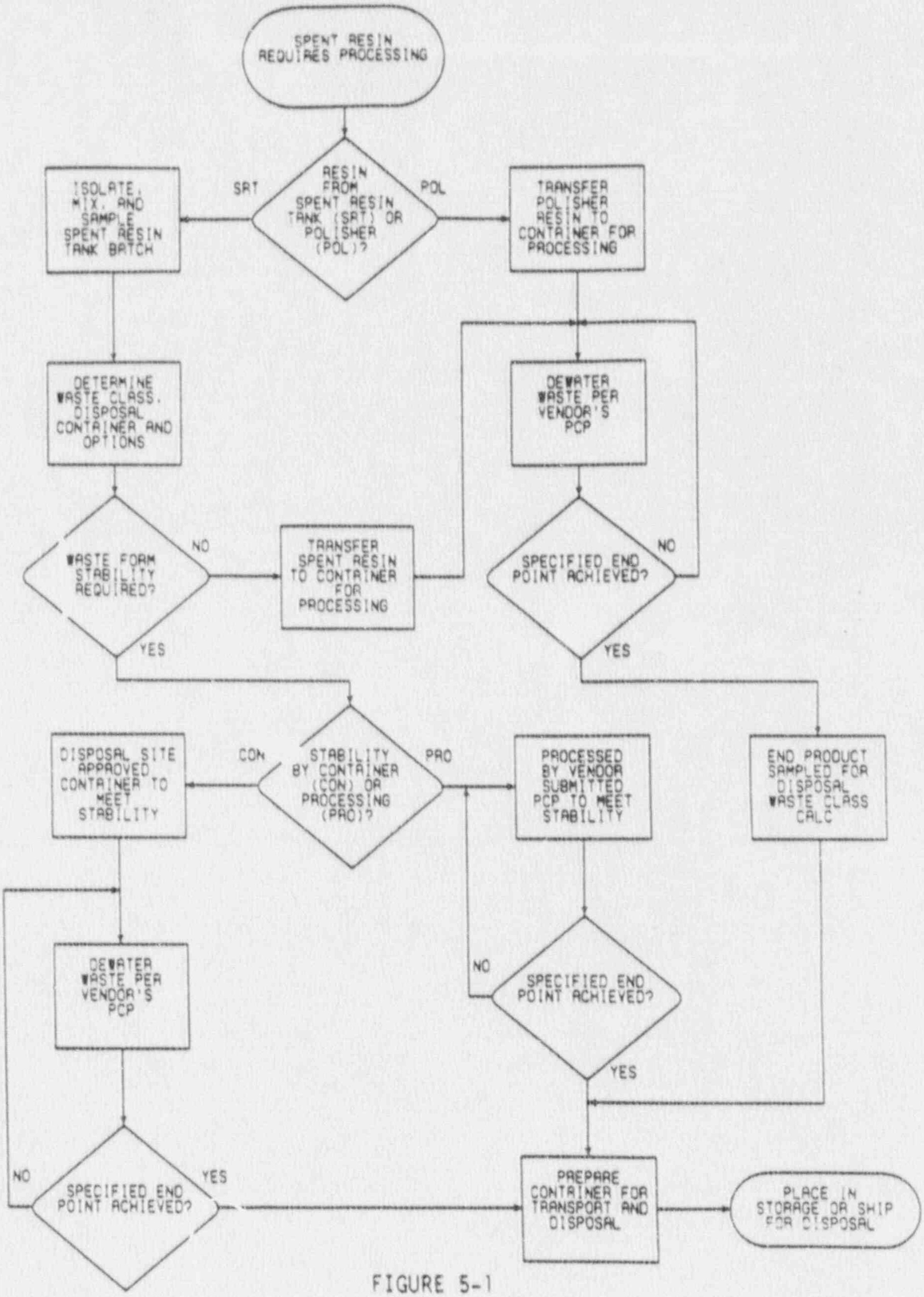

FIGURE 5-1

DEWATERING PROCESS FLOW CHART

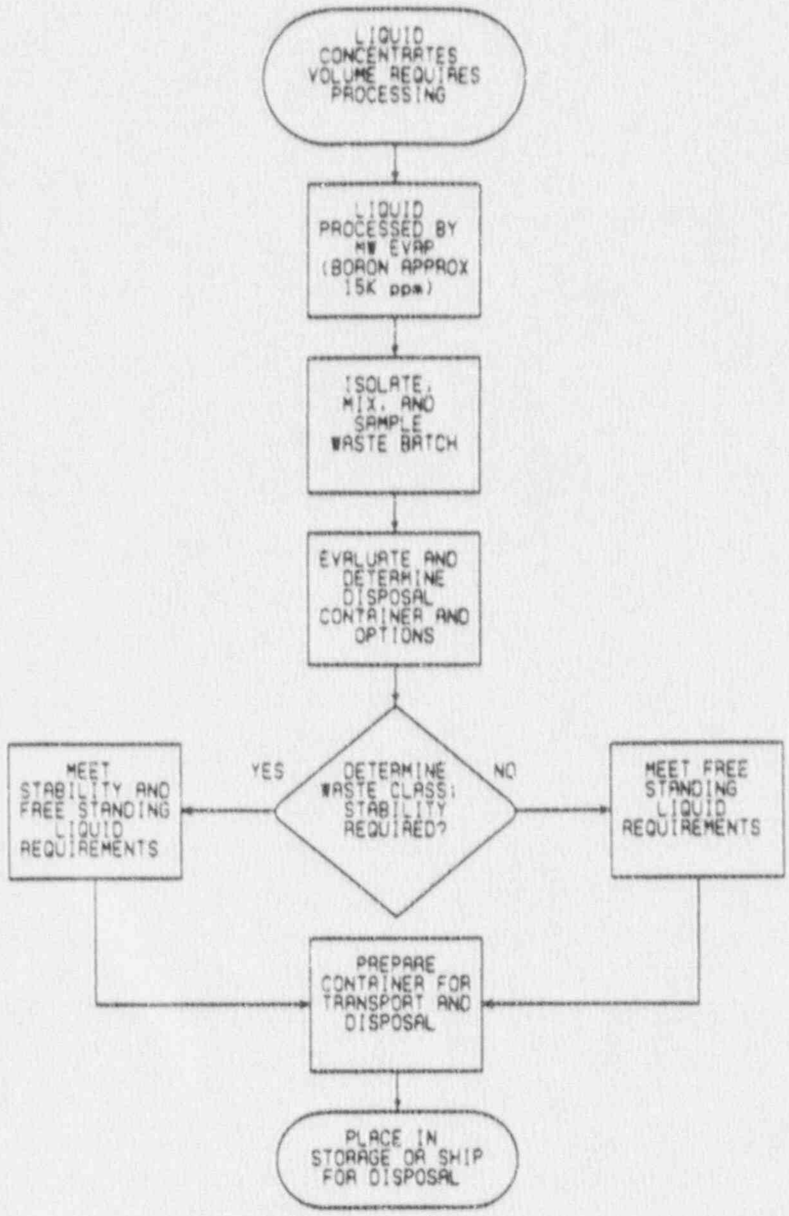

FIGURE 5-2

LIQUID PROCESS FLOW CHART

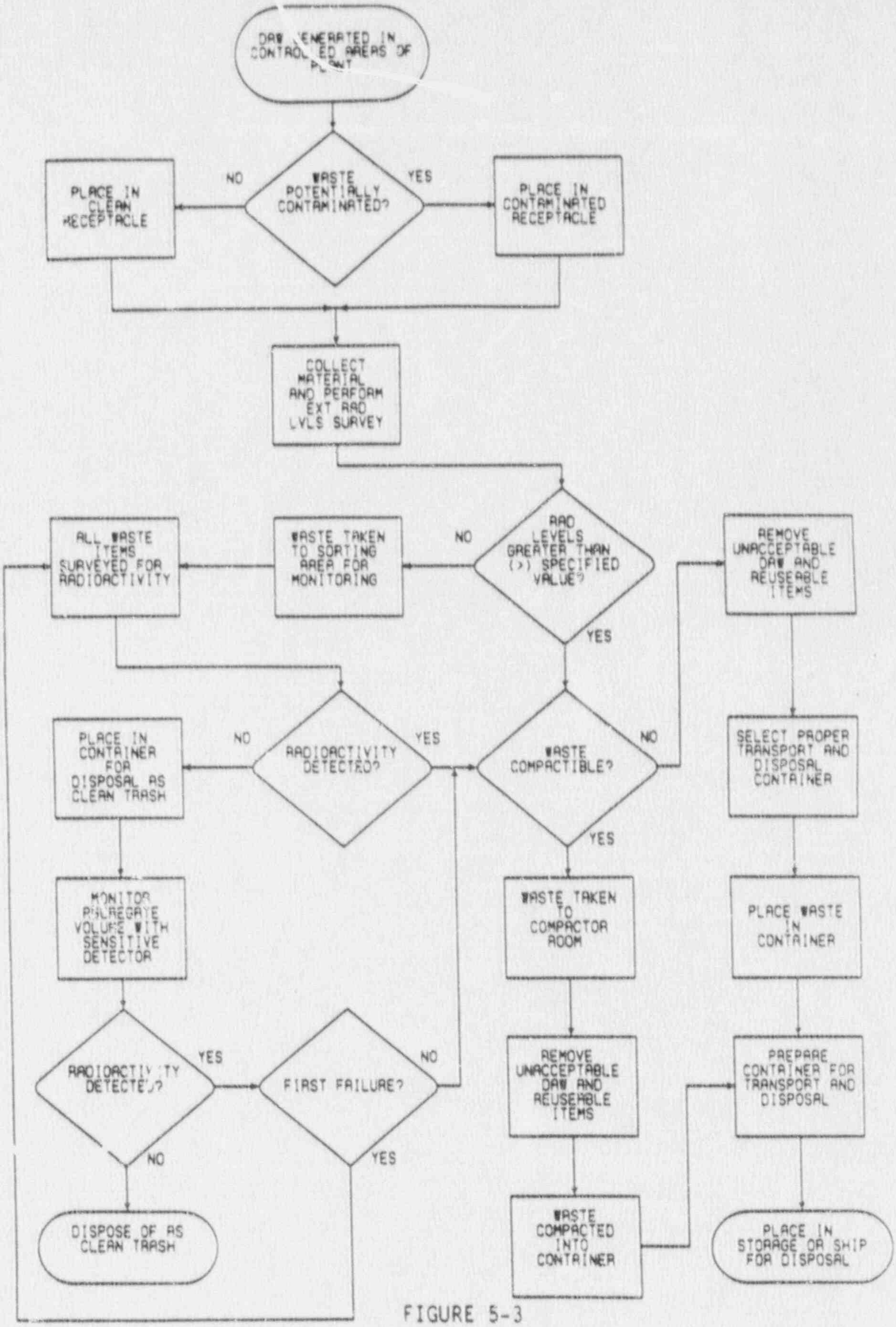

FIGURE 5-3

DAW PROCESS FLOW CHART

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 PROCEDURES

Activities associated with the implementation of the requirements of this program shall be conducted in accordance with approved station procedures or vendor documents and procedures that have been reviewed and approved per Sections 1.4 and 3.2.

6.2 QUALITY ASSURANCE

Quality Assurance related activities for radioactive waste processing are implemented as described in the Rancho Seco Quality Assurance Manual. Such activities include:

(1) Review of documents and procedures affecting the processing, packaging, handling, and transportation of radioactive waste.

(2) Review of procurement documents or services.

(3) Perform inspections as designated in applicable processing, packaging, and shipping procedures.

(4) Review applicable vendor QA programs for compliance with Regulatory and Rancho Seco requirements.

(5) Perform audits of the radioactive waste management program at least once per 24 months.

(NOTE: Technical Specification Administrative Control D6.5.3.m requires that audits of the Process Control Program and implementing procedures be performed at least once per 24 months.)

(6) Documentation and retention of documentation of waste processing, packaging and shipping activities.

These activities provide assurance that the final waste form, packaging, labeling, and transportation are in accordance with applicable regulations and requirements.

6.3 CHANGES TO THE PCP

Changes to this PCP shall be made in accordance with Technical Specification D6.13. Technical Specification D6.13 requires that changes to the PCP:

(1) Shall be documented and records of reviews performed shall be retained for the duration of the unit Operating License. This documentation shall contain:

(a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s), and

(b) A determination that the change will maintain the overall conformance of the waste product to existing requirements of Federal, State, or other applicable regulations.

(2) Shall become effective after review and acceptance by PRC and the approval of the AGM, Nuclear.

6.4 DOCUMENTATION

Procedures for radioactive waste PROCESSING, packaging, and transportation shall include requirements for maintaining and retaining LLW processing, packaging, and transportation records. Detailed records for each container of waste shall be maintained.

6.5 TRAINING

Rancho Seco and vendor personnel responsible for waste processing, packaging and transportation activities shall be trained and qualified to ensure that waste PROCESSING is performed in accordance with applicable requirements. Training programs shall establish a schedule for periodic requalification of at least once every two years. Rancho Seco personnel shall verify the training of vendor personnel.

SECTION 7.0

REFERENCES

7.1 Code of Federal Regulations:

- 7.1.1 Title 10, Parts 20, 61, and 71
- 7.1.2 Title 49, Part 173
- 7.1.3 Title 40, Part 261

7.2 USNRC, Low-Level Waste Licensing Branch, Technical Position on Radioactive Waste Classification, May 1983

7.3 USNRC, Low-Level Waste Licensing Branch, Technical Position on Waste Form, May 1983

7.4 USNRC, Guidelines for Preparation and Implementation of a Solid Waste Process Control Program (Proposed), Draft Revision 3, September 1986

7.5 USNRC, Office of Inspection and Enforcement, IE Information Notices:

- 7.5.1 IEN 79-09, "Spill of Radioactively Contaminated Resins"
- 7.5.2 IEN 83-14, "Dewatered Spent Ion Exchange Resin Susceptibility to Exothermic Chemical Reaction"
- 7.5.3 IEN 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities"
- 7.5.4 IEN 86-20, "Low-Level Radioactive Waste Scaling Factors, 10 CFR Part 61"
- 7.5.5 IEN 87-03, "Segregation of Hazardous and Low-Level Radioactive Wastes"
- 7.5.6 IEN 87-07, "Quality Control of Onsite Dewatering/ Solidification Operations By Outside Contractors"
- 7.5.7 IEN 88-08, "Chemical Reactions With Radioactive Waste Solidification Agents"
- 7.5.8 IEN 89-27, "Limitations on the Use of Waste Forms and High Integrity Containers for the Disposal of Low-Level Radioactive Waste"

7.6 Rancho Seco Technical Specifications, Sections D6.5.3.m and D6.13

7.7 Radwaste Control Manual

- 7.7.1 R.P.309.I.07, Waste Stream Identification and Sampling
- 7.7.2 R.P.309.II.01, Resin and Filter Media Dewatering using PNSI Nuclear Packaging Dewatering System
- 7.7.3 R.P.309.II.04, Solidification of Water or Resin using Cement with PNSI Radwaste Solidification System
- 7.7.4 R.P.309.II.09, Segregation and Free Release of Controlled Area Waste

7.8 Rancho Seco Quality Manual

7.9 Vendor Documents

7.9.1 TP-05, Radwaste Solidification System Topical Report

7.9.2 TP-02-P, Nuclear Packaging, Inc. Dewatering System Topical Report

7.10 RSAP-0303, Plant Modification

ATTACHMENT VII

Justification for Deletion of Appendix B

Proposed Amendment No. 102, Revision 1 and

Appendix C Related Justification

SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ P. O. Box 15830, Sacramento CA 95852-1830, (916) 452-3211
AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

SM 89-005

JUL 0 5 1989

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Docket No. 50-312
Rancho Seco Nuclear Generating Station
License No. DPR-54
PROPOSED AMENDMENT NO. 102, REVISION 1, RESUBMITTAL

Attention: George Knighton

In accordance with 10 CFR 50.90, the Sacramento Municipal Utility District proposes to amend Operating License DPR-54 for Rancho Seco and therefore resubmits Proposed Amendment No. 102, Revision 1. By letter to the Commission dated December 12, 1984, the District submitted the initial version of Proposed Amendment No. 102. Revision 1 of Proposed Amendment No. 102, submitted July 27, 1988, replaced the initial submittal in its entirety.

At the Commission's request, by letter dated May 10, 1989, the District withdrew three Proposed Amendments including No. 102, Revision 1. The amendments were withdrawn to allow the District and NRC to concentrate resources on the Improved Technical Specifications.

The shutdown of Rancho Seco on June 7, 1989, has made the submittal of Improved Technical Specifications impractical. The District is therefore resubmitting Proposed Amendment No. 102, Revision 1.

Pursuant to 10 CFR 50.91(b)(1), the Radiological Health Branch of the California State Department of Health Services has been informed of this proposed amendment by mailed copy of this submittal.

Since this is a resubmittal of Revision 1 to Proposed Amendment No. 102, no additional license fees are required.

Members of your staff with questions requiring additional information or clarification may contact Mr. Richard Mannheimer at (209) 333-2935, extension 4919.

FILE COPY

State of California
SS
County of Sacramento

Dan R. Keuter, being first duly sworn, deposes and says: that he is Assistant General Manager, Nuclear Plant Manager of Sacramento Municipal Utility District (SMUD), the licensee herein; that he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information, and belief, and that he is authorized to execute this document on behalf of said licensee.

Dan Keuter

Dan R. Keuter
Assistant General Manager
Nuclear Plant Manager

OFFICIAL SEAL
ELIZABETH L. GANDY
NOTARY PUBLIC-CALIFORNIA
SACRAMENTO COUNTY
MY COMM. EXP. NOV. 20, 1992

Subscribed and affirmed to before me on this 5th day of July, 1989.

Elizabeth L. Gandy

Elizabeth L. Gandy
Notary Public

Attachment

cc w/atch: A. D'Angelo, NRC, Rancho Seco
J. B. Martin, NRC, Walnut Creek
P. Szalinski, State of California
INPO
MIPC (2)

ATTACHMENT I

Description of Change, Reason for Change, and

No Significant Hazards Consideration for

Proposed Amendment No. 102, Revision 1

Safety Analysis Report

Description of Change

Proposed Amendment No. 102, Revision 1 deletes the remaining non-radiological items contained within Appendix B of the Rancho Seco Technical Specifications (Tech Specs). Pages i, ii, and 1 through 50 of Appendix B to the Tech Specs are removed and replaced by new Page i.

Reason for Change

The non-radiological items remaining in Appendix B which were not deleted by Amendment No. 45 are eliminated based on the following information. The Chemistry and Radiation Protection groups at Rancho Seco have been involved with a non-radiological environmental surveillance program for over 13 years. This program consists of erosion protection, draft contaminant monitoring, noise monitoring, and fogging patterns associated with the cooling towers. Samples obtained during the implementation of this program have not indicated any hazardous effects to the environment or the general public. Historical data shows that no severe erosion of the stream bed or degradation of soil banks surrounding the effluent stream has occurred. The only significant amount of erosion has taken place during the heavy winter and spring rains. This would have occurred regardless of plant operation. The deletion of noise surveys from Appendix B can be justified since historical data shows that the surveys have been within acceptable limits. Monitoring of fogging patterns can also be deleted. Past observations of fogging patterns associated with the cooling towers have shown no significant fog increase. The only fog observed has been normal valley fog during the winter months. For these reasons, the District intends to delete these non-radiological items from Appendix B of the Tech Specs.

The Administrative Controls Tech Specs in Section 5 of Appendix B are bounded by the administrative controls established in Section 6 of Appendix A. The need for administrative controls in Appendix B no longer exists.

No Significant Hazards Consideration

The changes proposed in this Tech Spec amendment have no significant impact on plant safety or on site personnel, or public health and safety. Past implementation of the non-radiological Tech Specs has not indicated any hazardous effects to the environment or the general public due to the operation of Rancho Seco. These non-radiological operating restrictions were originally imposed because acceptable operation was not yet demonstrated regarding the environmental impact of Rancho Seco. The District believes relief from the non-radiological items in Appendix B of the Tech Specs should be granted because the non-radiological impact of Rancho Seco has been demonstrated as acceptable under the criteria established by the Commission.

In accordance with Amendment No. 45, the NRC is provided with a copy of any changes to the National Pollutant Discharge Elimination System (NPDES) permit and any permit violations. Appendix B Environmental Tech Specs which pertain to non-radiological water quality requirements were deleted with Amendment No. 45. Water quality limits and monitoring programs associated with this permit are under the jurisdiction of the California Regional Water Quality Control Board. All non-radiological environmental monitoring program changes and violations are handled by the appropriate federal, state, local and regional authorities.

Amendment No. 53 added several Administrative Controls Tech Specs which duplicate or exceed requirements contained in Section 5.0 of Appendix B. A letter from John F. Stolz, Chief, Operating Reactors Branch #4, Division of Licensing dated September 11, 1985, asked for assurance that the administrative controls in Section 6 of the Appendix A Tech Specs duplicate or exceed the following items:

1. Appendix B, Administrative Controls, deletion of Sections 5.3.A.2 and 5.3.B.2, requirements to review onsite tests and experiments and results thereof when such tests have environmental significance.

2. Appendix B, Administrative Controls, deletion of Section 5.6.3, reporting requirements and evaluation of plant design changes when the changes may adversely impact the environment.

3. Appendix B, Administrative Controls, deletion of Section 5.7, requirement to maintain environmental records.

Appropriate replacement Tech Specs exist for item 1 in several sections. Deletion of Appendix B, Administrative Controls, Section 5.3.A.2 is justified because of the review responsibilities required of the Plant Review Committee (PRC) in Specification 6.5.1.6 of Appendix A. Part a. states that the PRC will review the safety evaluations of all procedures required by Specification 6.8. Specification 6.8 includes: Process Control Program implementation procedures, Offsite Dose Calculation Manual implementation procedures, Radiological Environmental Monitoring Program implementation procedures, and Effluent Control and Environmental Monitoring Quality Assurance Program procedures. The PRC reviews the safety evaluations of all proposed tests and experiments that affect nuclear safety, and the PRC also reviews facility operations to detect potential safety hazards.

Appendix B, Administrative Controls, Section 5.3.B.2 is no longer necessary because of Appendix A, Specification 6.5.2.6d. The Management Safety Review Committee (MSRC) reviews all changes to the Tech Specs or the Operating License. Tech Spec changes are all accompanied by a safety analysis evaluating the impact of the change.

The requirements of Administrative Controls Section 5.6.3 of Appendix B are satisfied in Section 6 of Appendix A. The NRC requested that Specifications 5.6.3.1 and 5.6.3.2 of Appendix B be retained in Appendix A. These two Specifications were included in the District's submittal of Proposed Amendment No. 138, Revision 2 and approved as part of Amendment No. 96 by NRC letter dated February 19, 1988. Per 10 CFR 50.90 and 10 CFR 50.4 all license amendments are required to be submitted to the NRC for approval. All proposed Tech Spec amendments are submitted with a Safety Analysis Report, which contains a No Significant Hazards Determination. Specification 5.6.3.3 in Appendix B is obsolete.

The requirements of Section 5.7 to Appendix B (Records Retention) are satisfied by Section 6.10 of Appendix A.

Based on the above discussion, the probability of occurrence or consequence of an accident previously evaluated in the SAR will not be increased, the possibility of occurrence of an accident of a different type than previously evaluated in the SAR will not be created, and the margin of safety as defined in the basis for any Tech Spec is not reduced; therefore, an Unreviewed Safety Question is not involved.

The District has reviewed the proposed changes against each of the criterion of 10 CFR 50.92 and concluded that the changes to the Tech Specs discussed above would not:

a. Significantly increase the probability or consequences of an accident previously evaluated because the proposed Tech Spec changes will not change the way any plant system or component important to safety is operated. The proposed amendment eliminates unnecessary Tech Spec requirements. Administrative controls deleted in Appendix B which were of concern to the Commission are duplicated or exceeded in Appendix A.

b. Create the possibility of a new or different type of accident than previously evaluated because implementation of the non-radiological Tech Specs is now considered to be unnecessary. Acceptable operation of Rancho Seco with regard to environmental impact has been demonstrated. All appropriate administrative controls exist in Appendix A of the Tech Specs.

c. Involve a significant reduction in the margin of safety because the deletion of the remaining Appendix B Tech Specs does not affect any system or component important to safety. The proposed changes do not impact the margin of safety defined in any basis to the Tech Specs. The non-radiological impact of Rancho Seco has been demonstrated acceptable under criteria established by the Commission.

On the basis of the above, the District concludes that the proposed changes do not constitute any significant hazard to the public, and in no way endanger the public's health and safety.

ATTACHMENT II

Technical Specification Pages Affected by

Proposed Amendment No. 102, Revision 1

Appendix B

Remove	Insert
i	i
ii	---
1 - 50	---

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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APPENDIX B

Appendix B has been deleted in its entirety.

Proposed
Amendment No. 45, 53, 98, 102, Revision 1

Safety Analysis and
No Significant Hazards Consideration for
the Deletion of Appendix B Based on
the Proposed Appendix C
Technical Specifications

SAFETY ANALYSIS REPORT

Description of Change

Proposed Amendment No. 182, Revision 2 (Appendix C) deletes, in part, the remaining non-radiological items contained within Appendix B of the Rancho Seco Technical Specifications (Tech Specs). Pages i, ii, and 1 through 50 of Appendix B to the Tech Specs are removed and replaced by new Page 1.

Reason for Change

The non-radiological items remaining in Appendix B which were not deleted by Amendment No. 45 are eliminated based on the following information. The Chemistry and Radiation Protection groups at Rancho Seco have been involved with a Non-Radiological Environmental Surveillance Program for over 15 years. This program consists of erosion protection, draft contaminant monitoring, noise monitoring, and fogging patterns associated with the cooling towers. Samples obtained during the implementation of this program have not indicated any hazardous effects to the environment or the general public. Historical data shows that no severe erosion of the stream bed or degradation of soil banks surrounding the effluent stream has occurred. The only significant amount of erosion has taken place during the heavy winter and spring rains. This would have occurred regardless of plant operation. The deletion of noise surveys from Appendix B can be justified since historical data shows that the surveys have been within acceptable limits. Monitoring of fogging patterns can also be deleted. Past observations of fogging patterns associated with the cooling towers have shown no significant fog increase. The only fog observed has been normal valley fog during the winter months. Because the plant is defueled further obviates the need for a Non-Radiological Environmental Surveillance Program in the PDM. For these reasons, the District intends to delete these non-radiological items from the Appendix B Tech Specs.

The Administrative Controls Tech Specs in Section 5 of Appendix B are bounded by the administrative controls established in Section 6 of Appendix C. The need for administrative controls in Appendix B no longer exists.

No Significant Hazards Consideration

The changes proposed in this Tech Spec amendment have no significant impact on plant safety or on site personnel, or public health and safety. Past implementation of the non-radiological Tech Specs has not indicated any hazardous effects to the environment or the general public due to the operation of Rancho Seco. These non-radiological operating restrictions were originally imposed because acceptable operation was not yet demonstrated regarding the environmental impact of Rancho Seco. The District believes relief from the non-radiological items in Appendix B of the Tech Specs should be granted because the non-radiological impact of Rancho Seco has been demonstrated as acceptable under the criteria established by the Commission.

In accordance with Amendment No. 45, the NRC is provided with a copy of any changes to the National Pollutant Discharge Elimination System (NPDES) permit and any permit violations. Appendix B Tech Specs which pertain to non-radiological water quality requirements were deleted with Amendment No. 45. Water quality limits and monitoring programs associated with this permit are under the jurisdiction of the California Regional Water Quality Control Board. All non-radiological environmental monitoring program changes and violations are handled by the appropriate federal, state, local and regional authorities.

Amendment No. 53 added several Administrative Controls Tech Specs to Appendix A which duplicate or exceed requirements contained in Section 5.0 of Appendix B. These requirements are included in Appendix C. A letter from John F. Stolz, Chief, Operating Reactors Branch #4, Division of Licensing dated September 11, 1985, asked for assurance that the Administrative Controls Tech Specs duplicate or exceed the following items:

1. Appendix B, Administrative Controls, Sections 5.3.A.2 and 5.3.B.2, requirements to review onsite tests and experiments and results thereof when such tests have environmental significance.

2. Appendix B, Administrative Controls, Section 5.6.3, reporting requirements and evaluation of plant design changes when the changes may adversely impact the environment.

3. Appendix B, Administrative Controls, Section 5.7, requirement to maintain environmental records.

Appropriate replacement Tech Specs exist for item 1 in several sections of Appendix C. Deletion of Appendix B, Administrative Controls, Section 5.3.A.2 is justified because of the review responsibilities required of the Nuclear Safety Review and Audit Committee (NSRAC) in Specification D6.5.1.7 of Appendix A. Part d. states that the NSRAC will review the safety evaluations of all procedures, plans, manuals, and programs required by Specification D6.8. Specification D6.8 includes: Process Control Program implementation, Offsite Dose Calculation Manual implementation, Radiological Environmental Monitoring Program implementation, and Effluent Control and Environmental Monitoring Quality Assurance Program implementation. The NSRAC reviews the safety evaluations of all proposed tests and experiments that affect Nuclear Safety, and they also review abnormal conditions or performance of equipment to detect potential safety hazards (Specifications D6.5.1.7e,f,g, and j).

Appendix B, Administrative Controls, Section 5.3.B.2 is no longer necessary because of Appendix C, Specification D6.5.1.7c. The NSRAC reviews all changes to the Tech Specs and the Facility Operating License. Tech Spec and License changes are accompanied by a safety analysis evaluating the impact of the change.

The environmental reporting requirements of Administrative Controls Section 5.6.3 of Appendix B are satisfied in Section 6 of Appendix C. The NRC requested that Specifications 5.6.3.1 and 5.6.3.2 of Appendix B be retained in the Tech Specs. These two Specifications are included in Appendix C as Specification D6.9.6.

Per 10 CFR 50.90 and 10 CFR 50.4 all license amendments are required to be submitted to the NRC for approval. Since proposed Tech Spec amendments are submitted with a Safety Analysis Report and a No Significant Hazards Determination, Specification 5.6.3.3 in Appendix B is obsolete and is not included in Appendix C.

The requirements of Section 5.7 to Appendix B (Records Retention) are satisfied by Section 6.10 of Appendix C.

Based on the above discussion, the probability of occurrence or consequence of an accident previously evaluated in the SAR will not be increased, the possibility of occurrence of an accident of a different type than previously evaluated in the SAR will not be created, and the margin of safety as defined in the basis for any Tech Spec is not reduced; therefore, an Unreviewed Safety Question is not involved.

The District has reviewed the proposed changes against each of the criterion of 10 CFR 50.92 and concluded that the changes to the Tech Specs discussed above would not:

a. Significantly increase the probability or consequences of an accident previously evaluated because the proposed Tech Spec changes will not change the way any plant system or component important to safety is operated. The proposed amendment eliminates unnecessary Tech Spec requirements. Administrative controls deleted in Appendix B which were of concern to the Commission are duplicated or exceeded in Appendix C.

b. Create the possibility of a new or different type of accident than previously evaluated because implementation of the Appendix B Tech Specs in the PDM is considered unnecessary. Acceptable operation of Rancho Seco with regard to environmental impact has been demonstrated. Appropriate administrative controls are included in Appendix C.

c. Involve a significant reduction in the margin of safety because the deletion of the remaining Appendix B Tech Specs does not affect any system or component important to safety. The proposed changes do not impact the margin of safety defined in any basis to the Tech Specs. The non-radiological impact of Rancho Seco has been demonstrated acceptable under criteria established by the Commission.

On the basis of the above, the District concludes that the proposed change does not constitute a significant hazard to the public or endanger the health and safety of the public.

Appendix B Technical Specification Pages Affected by

Proposed Amendment No. 182, Revision 2

Appendix B

Remove	Insert
i	i
ii	---
1 - 50	---

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

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APPENDIX B

Appendix B has been deleted in its entirety.

ATTACHMENT VIII

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

REVISION SUMMARY:

Complete rewrite to incorporate the provisions of USNRC Generic Letter 89-01 in support of submittal of Technical Specification Proposed Amendment No. PA-182 (Appendix C Defueled Technical Specifications).

This revision is administrative in that no REMP requirements (Table 1) or sample locations (Table 6) have been deleted. Three editorial changes have been made to Table 6 to correct previous omissions.

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0.0 POLICY

The Sacramento Municipal Utility District (SMUD) and the Rancho Seco Nuclear Generating Station recognize their responsibility to comply with the Technical Specifications and the applicable regulations, codes, standards and industry-wide criteria for establishing and maintaining a viable Radiological Environmental Monitoring Program. We are committed to operating the Rancho Seco Nuclear Generating Station in such a manner that will assure proper radiation protection to all employees, contractors and the general public. To this end, we have committed to performing an environmental sampling program which meets the intent of the applicable regulations while providing an accurate assessment of the radiological environment in and around the environs of the Rancho Seco site.

1.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM BASES

The Sacramento Municipal Utility District and the Rancho Seco Nuclear Generating Station have instituted a Radiological Environmental Monitoring Program (REMP) which this manual serves to implement. The REMP is based upon the information contained in Title 10 of the Code of Federal Regulations, Part 20, Section 106 (10 CFR 20.106). That Regulatory basis and associated guidelines have been the foundation of the REMP and its programmatic elements which:

1. Provide the technological basis of, and the instruction for, monitoring the site and environs for radioactivity of all sources, including:

   a. naturally occurring background
   b. releases during normal operations
   c. operational occurrences and postulated accidents
   d. weapons testing and major nuclear accidents which contribute to detectable radioactivity in the environs.

2. Provide the means to verify the effluent control program of the Rancho Seco Nuclear Generating Station.

3. Meet minimum limits for detecting radioactive elements in samples collected from the environs, or direct measurements in the field.

4. Provide measurements of radiation and radioactive materials in those exposure pathways, i.e., liquid, gaseous, and direct radiation, and for those radionuclides, i.e., iodine, cesium, and cobalt, which lead to the highest potential radiation exposure of individuals resulting from station operation.

This Manual contains the minimum requirements for the conduct of the Rancho Seco Radiological Environmental Monitoring Program (REMP). The requirements are consistent with USNRC regulations formerly contained in the Rancho Seco Appendix A Technical Specifications, through Amendment #114, as Limiting Conditions for Operation, Surveillance Standards (and associated Bases) and Administrative Controls.

This revision was prompted by the provisions of Generic Letter 89-01 (Reference 9.13). For ease of intercomparison, each section contains a reference to the applicable Technical Specification (TS) Appendix A section from which the requirements were derived.

2.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM DESCRIPTION

The Radiological Environmental Monitoring Program is under the cognizance of the Assistant General Manager, Nuclear, with the responsibility for the administration and oversight of the program assigned to the Manager, Nuclear Quality Assurance.

The design of the program is consistent with the intent of Title 10 Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation" Section 106. To implement these requirements, the Technical Specifications, Offsite Dose Calculation Manual, Health Physics Implementation Procedures and Surveillance Procedures have been developed. The implementing procedures address specific areas in the program that require direct attention for completion.

Revision 3 to the REMP Manual was prompted by the REMP Action Plan which was implemented during August, 1989.

2.1 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM PARAMETERS

The monitoring and sampling aspects of the program are:

Identification of the effluent release pathways,

Identification of the human exposure pathways,

Identification of the land usage parameters by the population within a ten mile radius of the site.

Three principal release pathways at Rancho Seco Nuclear Generating Station, are:

<u>Gaseous Effluents</u>	discharges from the reactor building stack, auxiliary building stack, and the auxiliary grade level vent.
<u>Liquid Effluents</u>	discharges which are released from the retention basins via the waste water disposal system [regenerant hold up tanks (RHUT) A and B].
<u>Direct Radiation</u>	radiation that emanates from radioactive material contained within tanks or other containers which are within the site boundary to humans outside of the site boundary.

The pathways to human exposure to radioactive materials in the effluent release pathways from Rancho Seco are:

<u>Gaseous</u> Inhalation of airborne radioactive material by humans, or by animals that inhale and retain the material in animal products eaten by the public, i.e., meat or milk.

- Consumption of radioactive particulate material which, although carried by air currents, is deposited onto or is taken up by water sources or plants consumed by humans, or by animals that provide products that are consumed by humans, i.e., milk or meat.

- Exposure from being immersed in air containing radioactive materials as a gas and/or particulates.

- Exposure to the direct radiation from radioactive materials that have been deposited onto surfaces from airborne releases.

<u>Liquid</u> Drinking of water from the release pathway by humans, or by animals that are a food source for humans.

- The consumption of fish, shellfish or other animals that have eaten fish or shellfish taken from water within the liquid release pathway.

- The consumption of animal meats or products of animals that have eaten vegetation that have been irrigated with water from the release pathway.

- The consumption by humans of fruit or vegetation grown in soil irrigated with water from the release pathway.

<u>Direct Radiation</u> The exposure to radiation emitted from radioactive materials within the Rancho Seco site boundary. Sources include, but are not limited to, the Borated Water Storage Tank, Reactor Coolant Storage Tank, and the Radioactive Waste Storage Area.

- The exposure from being immersed in the release pathway water, to radiation emanating from material contained in the water.

2.2 ANALYSIS OF EXPOSURE PATHWAYS

Exposure pathways are analyzed through a systematic process which identifies a sample medium or organism that is found in the effluent pathways. Usage factors are determined that will suitably represent biological concentration, retention or uptake which may ultimately represent a contribution to human exposure. The pathways to human exposure are evaluated through the analysis of data obtained from the performance of a land use census. The performance of the land use census is required by the Technical Specifications Section D6.8.3.b. The analysis of the effluent and exposure pathways enables the selection of sampling and monitoring locations that fall into one of two classes, those which are, and those which are not, influenced by effluent pathways. Those in the pathways are referred to as indicator locations. Several of the unaffected locations are selected to represent baseline or control locations.

Indicator locations provide data from the surrounding environment that may be influenced by the operation of the plant because they are nearby, downwind or downstream in the release pathway. Such data can be used to calculate doses to humans to verify compliance with 40 CFR 190, using methodology contained in the ODCM. (This is referred to as the MEMBER OF THE PUBLIC. The MEMBER OF THE PUBLIC is defined as any person who participates in activities that result in that person being in the actual pathways for offsite dose. A MEMBER OF THE PUBLIC who, based upon the land use census, is expected to receive the maximum offsite dose to real individuals, may be used to calculate doses to demonstrate compliance with 40 CFR 190.)

Control sample locations are to provide data that should not be influenced by the operations of Rancho Seco. These locations are selected based upon the distance from the plant, being upwind, or upstream of the release pathways. Data from these locations help discriminate between Rancho Seco releases and other natural or manmade events that may impact human exposure.

At Rancho Seco, potentially radioactive liquid effluent is discharged into Clay Creek. Continuously, an average flow of 8,500 gallons per minute of non-radioactive water is released above the discharge point. The continuous minimum flow and the liquid effluent release are the major effluent release pathway, and thus the exposure pathway for the station during normal operations. Prior to the minimum release rate being established, Clay Creek was a seasonal stream, formed as the confluence of three and one half square miles of drainage runoff upstream of the site. The now continuous flow of Clay Creek intersects Hadselville Creek North and West of California State Highway 104. Hadselville Creek intersects Laguna Creek just East of the Folsom Canal. Laguna Creek flows into the Cosumnes River approximately 20 miles from Rancho Seco. Hadselville and Laguna Creeks are also seasonal streams and also receive irrigation runoff during periods when irrigation is used. Because these streams are the major release pathway for liquid effluents from the site, the majority of recent program enhancements have focused on this effluent pathway.

The gaseous pathway analysis is also related to the land use census. This pathway is not confined by creek banks, but is subject to the meteorological conditions during the time of the release. This presents the requirements for having indicator and control sampling stations more evenly distributed with particular attention to those areas of greater population density. While not the major release nor exposure pathway, recent improvements in monitoring this pathway have been instituted.

The direct radiation exposure pathway is the least likely pathway for the exposure to plant radiation by humans. It is the most easily measured with the use of thermoluminescence dosimeters, which monitor continuously and passively. The dose is integrated over three months to accumulate a statistically significant exposure. The vast majority of the dose integrated by these detectors is delivered from primordial elements in the geological surface of the Earth, which contain naturally radioactive elements. A smaller fraction of the dose is delivered by cosmic radiation which has penetrated the Earth's atmosphere.

3.0 RADIOLOGICAL ENVIRONMENTAL MONITORING (TS 4.1, 3.22/4.26)

The REMP shall be conducted AT ALL TIMES as sp   ed in **Table 1**.

3.1 With the REMP not being conducted as specified in **Table 1**, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report (AREOR) required by Section 8.1, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions or seasonal unavailability.

3.2 With the level of radioactivity in an environmental sampling medium exceeding the Reporting Level of **Table 3** when averaged over any calendar quarter, in addition to complying the the requirements of Section 5.0, FUEL CYCLE DOSE, 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 Reporting Levels to be exceeded. This report will define corrective actions to reduce emissions such that potential exposures will meet 10 CFR 50 Appendix I annual dose limitations. When more than one of the radionuclides in **Table 3** are detected in the sampling medium, the Special Report shall be submitted if the Reporting Level fraction summation equals or exceeds unity (1.0).

When radionuclides other than those in **Table 3** are detected and are the result of plant effluents, this Special Report shall be submitted if the potential annual dose to an individual is greater than or equal to the calendar year limits of 10 CFR 50 Appendix I. This Special 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 AREOR.

3.3 With milk or fresh leafy vegetation samples unavailable from any of the sample locations required by **Table 1**, identify the cause of the unavailability of samples and the locations for obtaining replacement samples in the next AREOR. The locations from which samples were unavailable may then be deleted from **Table 6** provided the locations from which the replacement samples were obtained are added to **Table 6** as replacement locations, if available.

3.4 The radiological environmental monitoring samples shall be collected per **Table 1** from the locations shown in **Table 6**. These samples shall be analyzed to the requirements of **Table 1** and **Table 2**.

3.5 The environmental air monitors used for sampling the **Table 1** AIRBORNE EXPOSURE PATHWAY shall be subject to a MONTHLY function check and shall be calibrated ONCE EVERY 18 MONTHS.

3.6 The REMP required by Section 1.0 provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the Station operation. This monitoring program thereby implements Section IV.B.2 of Appendix I to 10 CFR 50 and supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and Offsite Dose Calculation Manual (ODCM) modeling of the environmental exposure pathways.

Guidance for Section 3.0 was provided by Reference 9.12. REMP changes may be initiated based on operational experience and changes in the regional population or agricultural practices. The detection capabilities required by **Table 2** are state of the art for routine environmental measurements in industrial laboratories. The LLDs for drinking water meet the requirements of 40 CFR 141.

4.0 LAND USE CENSUS (TS 3.23/4.27)

A Land Use Census shall be conducted annually and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden of greater than 500 square feet producing fresh leafy vegetation in each of the 16 meteorological sectors within a distance of five (5) miles. Broad leaf vegetation sampling may be performed at the Station site boundary in the direction sector with the highest deposition parameter in lieu of the garden census.

The Land Use Census shall also include information relevant to the liquid effluent pathway and gaseous effluent pathway such that the ODCM and the REMP Manual can be kept current with existing environmental and societal use of land surrounding the Station.

4.1 The Land Use Census shall be conducted by using methods that will provide the best results, such as door-to-door survey, aerial survey, or by consulting local agriculture authorities. The Land Use Census, or portions thereof, shall be conducted during the appropriate time of the year to provide the best results. The results of the Land Use Census shall be included in the AREOR covering the censused year as required by Section 8.1.2.

4.2 With the Land Use Census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in the ODCM for compliance with 10 CFR 50 Appendix I, identify the new locations in the next AREOR.

4.3 With the Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Section 3.0, Radiological Environmental Monitoring, add the new location(s) to Table 6 within 30 days or submit a Special Report to the Commission that identifies the cause(s) for exceeding these requirements and the proposed corrective actions for precluding recurrence.

The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s) [via the same exposure pathway] may be deleted from Table 6 after October 31 of the census year. Identify the new location(s) in the next AREOR including a revised figure(s) and table for the REMP Manual reflecting the new location(s).

4.4 The Section 4.0 requirements are provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the REMP and the ODCM are made if required by the results of the Land Use Census. These requirements also satisfy the requirements of Section IV.B.3 of Appendix I to 10 CFR 50.

Restricting the Land Use Census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetation consumption will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetation assumed (reference 7.15) to be consumed by a child. In specifying this minimum garden size, it was further assumed that 20 percent of the garden was used for growing broad leaf vegetation (e.g., lettuce or cabbage) and that the productivity was two (2) kg/m^2.

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

5.0 <u>FUEL CYCLE DOSE</u> (TS 3.25; TS 4.29 does not apply to the REMP)

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

5.1 With any of the Reporting Levels of **Table 3** being exceeded, calculations shall be made to determine whether the Section 5.0 fuel cycle dose/dose commitment limits have been exceeded. Contributions from direct radiation sources (including outside storage tanks, etc.) shall be included in this calculation.

5.2 If the Section 5.0 limits have been exceeded, prepare and submit to the Commission within 30 days a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the Section 5.0 limits. This Special Report shall also include a schedule for achieving conformance with the Section 5.0 limits.

This Special Report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, in a calendar year that includes the release(s) covered by this Special Report. This Special Report shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations.

5.3 If the estimated dose(s) exceeds Section 5.0 limits, and if the release condition resulting in the violation of 40 CFR 190 has not already been corrected, the Special Report shall also include a request for a variance in accordance with the provision of 40 CFR 190. Submittal of the Special Report is considered a timely request, and a variance is granted until USNRC staff action on the request is complete.

5.4 The Section 5.0 requirements are provided, in part, to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. For the Rancho Seco site, it is unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR 190 if the Station remains within twice the numerical guides for design objectives of 10 CFR 50 Appendix I and if direct radiation is kept small.

The Special Report will describe a course of action which should result in the limitation of the dose to a MEMBER OF THE PUBLIC for a calendar year to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of five (5) miles must be considered.

If the dose to any MEMBER OF THE PUBLIC is evaluated to exceed the requirements of 40 CFR 190, the Special Report along with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected) is considered to be a timely request and fulfills the requirements of 40 CFR 190 until USNRC staff action is completed.

An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation which is part of the uranium fuel cycle.

6.0 <u>INTERLABORATORY COMPARISON PROGRAM</u> (TS 3.26/4.30)

The laboratory performing analysis of Table 6 samples pursuant to the requirements of Table 1 shall AT ALL TIMES participate in an Interlaboratory Comparison Program (ICP) approved by the Commission.

6.1 With ICP analyses not being performed as required in Section 6.0, report the corrective actions taken to prevent a recurrence to the Commission in the AREOR as required by Section 8.1.

6.2 A summary of the results obtained as a participant in the ICP shall be included in the AREOR as required by Section 8.1.

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

7.0 <u>DEFINITIONS</u> (TS 3.0)

7.1 FORTNIGHTLY - Once per fourteen (14) days

7.2 RESTRICTED AREA - That portion of the Station property, access to which is controlled by security fencing, equipment and personnel.

7.3 SITE BOUNDARY - That line beyond which the land is neither owned, nor leased, nor otherwise controlled by the District.

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

8.0 <u>RADIOLOGICAL REPORT REQUIREMENTS</u> (TS 6.9.2)

8.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (AREOR)

8.1.1 An AREOR covering the operation of the Station during the previous calendar year shall be submitted to the USNRC prior to May 1 of each year.

8.1.2 The AREOR shall include summaries and statistical evaluations of the results of the radiological environmental surveillance activities for the report period, including (as appropriate) a comparison with operational controls. The AREOR shall also include the results of the Land Use Census required by Section 4.0, LAND USE CENSUS. In the event a radionuclide concentration should be confirmed in excess of the Reporting Level in **Table 3** by environmental measurements, the AREOR shall describe a planned course of corrective action.

8.1.3 The AREOR shall include summarized and tabulated results of all radiological environmental samples taken during the AREOR period. In the event that some results are not available for inclusion, the AREOR 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.

8.1.4 The AREOR shall include a summary description of the REMP (including a map of all sampling locations keyed to a table giving distances and directions from the Reactor Building) and the results of participation in the Interlaboratory Comparison Program required by Section 6.0. The AREOR shall also include information related to Section 5.0, Fuel Cycle Dose.

8.2 SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (SARERR)

Any changes made to the REMP MANUAL during the SARERR reporting period shall be included in that SARERR.

9.0 REFERENCES

The following documents pertain to the design and conduct of radiological environmental monitoring programs:

9.1 American National Standards Institute (ANSI), Performance, Testing and Procedural Specifications for Thermoluminescence Dosimetry (Environmental Applications), ANSI Standard N545 (1975).

9.2 American Nuclear Insurers and Mutual Atomic Energy Liability Underwriters (ANI/MAELU), Environmental Monitoring Programs Information Bulletin 86-1 (1986).

9.3 ANI/MAELU, "Radiological Environmental Monitoring," Engineering Inspection Criteria for Nuclear Liability Insurance, Section 5.2, Revision 2 (1988).

9.4 ANI/MAELU, Nuclear Liability Insurance Records Retention, Information Bulletin 80-1A, Rev. 2 (1986).

9.5 Committee on the Biological Effects of Ionizing Radiations (BEIR), The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980, BEIR III Report (1980).

9.6 National Council on Radiation Protection (NCRP), A Handbook of Radioactivity Measurements Procedures, NCRP Report No. 58, Second Edition (1985).

9.7 NCRP, Radiological Assessment: Predicting the Transport, Bioaccumulation and Uptake by Man of Radionuclides Released to the Environment, NCRP Report No. 76 (1984).

9.8 USEPA, "Environmental Standards for the Uranium Fuel Cycle," 40 CFR 190, Subpart B (1987).

9.9 USEPA, Upgrading Environmental Radiation Data, Health Physics Society Committee Report HPSR-1, EPA 520/1-80-012 (1980).

9.10 USNRC, "Criterion 64 - Monitoring Radioactive Releases," 10 CFR 50, Appendix A (1988).

9.11 USNRC, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material In Light Water Cooled Nuclear Power Reactor Effluents," 10 CFR 50, Appendix I (1988).

9.12 USNRC, "An Acceptable Radiological Environmental Monitoring Program," Branch Technical Position, Rev. 1 (November 1979).

9.13 USNRC, "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or the Process Control Program", Generic Letter 89-01 (January 31, 1989).

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

9.15 USNRC, Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, Regulatory Guide 1.113 (1977).

9.16 USNRC, Measuring and Reporting of Radioactivity in the Environs of Nuclear Power Plants, Regulatory Guide 4.1 (1973).

9.17 USNRC, Preparation of Environmental Reports for Nuclear Power Stations, Regulatory Guide 4.2, Rev. 2 (1976).

9.18 USNRC, Performance, Testing and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications," Regulatory Guide 4.13.

9.19 USNRC, Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment, Regulatory Guide 4.15, Rev. 1 (1979).

9.20 USNRC, Radiological Assessment: A Textbook on Environmental Dose Assessment, NUREG/CR-3332 (1983).

9.21 USNRC, Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements, NUREG/CR-4007 (1984).

9.22 USNRC, Radiological Effluent Technical Specifications for PWRs, NUREG-0472, Rev. 2 (July 1979).

9.23 USNRC, Radiological Monitoring by NRC Licensees for Routine Operations of Nuclear Facilities, NUREG-0475 (1978).

9.24 USNRC, Methods for Demonstrating LWR Compliance With the EPA Uranium Fuel Cycle Standard (40 CFR 190), NUREG-0543 (1980).

9.25 USNRC, "Permissible Levels of Radiation in Unrestricted Areas," 10 CFR 20.105 (1988).

9.26 USNRC, "Reports of Overexposures and Excessive Levels and Concentrations," 10 CFR 20.405 (1988).

9.27 Merril Eisenbud, Environmental Radioactivity From Natural, Industrial, and Military Sources, Third Edition (1987).

9.28 Rancho Seco Permanently Defueled Technical Specifications D1.10; D6.5.1.7.d; D6.5.4.k, l, n; D6.8.1.i, j; D6.8.3.b; D6.9.2.3; D6.10.2.m; D6.10.o; D6.14.2, 3.

10.0 IDENTIFICATION CONVENTION FOR TABLE 6 SAMPLE LOCATIONS

Sampling and monitoring sites designated in **Table 6** are identified using the following convention:

10.1 To establish the fact that the **Table 6** samples originate from the Rancho Seco REMP, the letter "R" precedes every sample site designator.

10.2 The next two (2) letters are selected to identify SAMPLE TYPE. Refer to **Table 4** for a listing of the SAMPLE CLASSES/TYPES and the associated two letter abbreviation.

10.3 The numbers following the SAMPLE TYPE abbreviation reflect the straight-line DISTANCE (miles) to the sample site, referenced to the center of the Reactor Building.

10.4 Following the distance, a SECTOR DESIGNATOR letter is included to specify which of the 16 meteorological sectors the sample site is encompassed. Refer to **Table 5** for a listing of the sector designators.

10.5 The final character in the sample site designation is the letter "O" which designates the sample as being one added to the REMP following Station initial criticality.

10.6 The present identification convention has been selected in preference to the system originally used to identify samples and sites. Since it is desireable to retain the ability to identify, and continue to use data from, previously collected samples, the former identification convention is also shown parenthetically in **Table 6**.

11.0 REPORTING RESULTS OF RADIOLOGICAL ENVIRONMENTAL DATA

The requirements for reporting radiological environmental data are specified in Section 8.0 of this manual. Those subsections which require supporting data from the Radiological Environmental Monitoring Program address the Annual Radiological Environmental Operating Report and the Semi-annual Radioactive Effluent Release Report. Special Reports are made specific in HPIP-2050, Radiological Environmental Monitoring Program Reports. Specified therein are conditions requiring special reports, and reporting requirements in days for submittal. This includes those calculations to provide rapid assurance of the degree of compliance with 10 CFR 50 Appendix I, and 40 CFR 190 calculations after releases of any origin.

12.0 SELECTION OF RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

In conjunction with the data base established from the land use census, the requirements of the Technical Specifications, and the guidance described in Section 2.0 of this Manual, the selection of sampling and monitoring sites was performed. These selected locations provide at least the minimum number of locations specified in **Table 1**.

Data was gathered from the land use census, Lawrence Livermore National Laboratory Rancho Seco Study Reports, Oak Ridge National Laboratory Study Reports, and from additional sampling sites from which materials have been collected. The information gathered was used to determine indicator sites. Presently, a sufficient number of control sites have been selected and are not anticipated to be increased in number.

Environmental thermoluminescence dosimeters are placed in the environs around the site. These devices passively monitor radiation in the immediate environs. Data from TLDs is trended to establish variations which are influenced by seasonal, meteorological, local and global sources. TLDs will also respond to radiation in the effluents of the plant if they pass in near proximity. The data is included in each quarterly environmental report.

Sample locations for the collection of the flora and fauna are concentrated in the liquid effluent pathway to the West. Representative samples of all the pathways and suitable locations are established in all directions. Air samplers are distributed to achieve a sampling of air from major wind directions across the site.

The Radiological Environmental Monitoring Program maintains at least those minimum sampling locations and type of samples to meet the requirements listed in **Table 1**. Many sample types and locations have been added to enhance characterizing the radiological environmental impact of plant operations. The increased sample locations and types in this manual will be maintained unless it is determined that they are no longer useful or necessary data sites. The number and type will not be reduced below those stated in **Table 1**.

Two special sites that have been established are vegetable gardens maintained by site personnel. One is established at the site boundary alongside the Clay Creek, and irrigated with water from the effluent stream. These data are considered essential for comparisons to vegetation not irrigated with effluent stream water for determination of bioaccumulation for soil types common to the environs. The second garden is at the North of the site, and is irrigated with domestic water. Unwashed samples will be taken from this garden to evaluate possible airborne materials to the North of the site via the gaseous effluent exposure pathway.

All of the current environmental sample locations required by Technical Specifications for the Radiological Environmental Monitoring Program are designated in **Table 6**. Detailed maps on which the sampling and monitoring locations are marked are maintained by the Environmental Monitoring Group.

Table 1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine	8	Continuous operation of sampler with sample collection as required by dust loading but at least once per week.	Radioiodine canister. Analyze at least once weekly for I-131. Particulate sampler. Analyze for Gross Beta radioactivity at least 24 hours following filter change. Perform gamma isotopic analysis on each sample where gross beta activity is greater than 10 times the yearly mean of control samples for the same sample period. Perform gamma isotopic analysis on composite (by location) for particulate filters sample at least once per quarter.
2. DIRECT RADIATION	At least 40 locations with 2 dosimeters at each location.	At least once per quarter.	Gamma dose. At least once per quarter.

* Sample locations are shown in Table 6.

Table 1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

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

* Sample locations are shown in Table 6.

Table 1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples*	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	4	At least weekly when animals are on pasture; at least once per month. at other times.	Gamma isotopic analysis and I-131 analysis of each sample.
b. Fish and Invertebrates	3	At least quarterly. One sample of each species as listed in Table 6.	Gamma isotopic analysis of edible portion of each sample.
c. Food	4	At time of harvest. One sample of each of the several classes of food products as shown in Table 6.	Gamma isotopic analysis on edible portion of each sample.

* Sample locations are shown in Table 6.

Table 2

MAXIMUM VALUES FOR THE LOWER LIMIT OF DETECTION, LLD[a,d]

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg-wet)	Milk (pCi/l)	Food Products (pCi/kg-wet)	Mud and Silt (pCi/kg-wet)
Gross Beta	4[b]	0.01				
H-3	2,000(1,000[b])					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			150
Co-60	15		130			150
Zn-65	30		260			
Zr,Nb-95	15[e]					
I-131	1[b]	0.07		1	60	
Cs-134	15(10[b])	0.01[c]	130	15	60	150
Cs-137	18(10[b])	0.01[c]	150	18	80	180
Ba,La-140	15[e]			15[e]		

Table 2 (Continued)

MAXIMUM VALUES FOR THE LOWER LIMIT OF DETECTION, LLD[a,d]

a The Low Limit of Detection (LLD) values for the radionuclides presented in Table 2 are those recommended in Reference 9.12.

The LLD of a radioanalysis system is that value which will indicate the presence or absence of radioactivity in a sample when the probability of a false position and of a false negative determination is stated. The probabilities of the false positive and false negative determinations are taken as equal to 0.05. The equation for estimating the maximum LLD is given by the following equation:

$$LLD = \frac{2.71/t_s + 3.29S_b}{3.7E\text{-}2\ (YEV)\exp(-\lambda t_c)},\ pCi/l,\ pCi/kg\text{-wet or } pCi/m^3$$

where:

2.71 = factor to account for Poisson statistics at very low background count rate

3.29 = twice the constant used to establish the one-sided 0.95 confidence interval

S_b = standard deviation of the background count rate

= $[B/(t_b t_s) + B/t_b^2]^{0.5}$

B = background counts

t_b = background count interval, sec

t_s = sample count interval, sec

3.7E-2 = conversion factor, dis/sec/pCi

Table 2 (Continued)

MAXIMUM VALUES FOR THE LOWER LIMIT OF DETECTION, LLD[a,d]

Y = radiochemical process yield (product of all factors such as abundance, chemical yield, etc.)

E = counting efficiency, cts/dis

V = sample volume or mass, l or kg

λ = radioactive decay constant for the associate nuclide, sec^{-1}

t_c = elapsed time from the midpoint of sample collection to the midpoint of counting, sec

The LLD is defined as an a priori (before the fact) estimate and is not to be calculated for each sample analyzed on an a posteriori (after the fact) basis.

Occasionally, unavoidably small sample sizes or other uncontrollable circumstances may result in a priori LLD values not being met. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

b LLD for water samples utilized for human consumption only.

c Composite analysis LLD is shown; individual sample LLD is 0.05 pCi/m^3.

d Other peaks which are measurable and identifiable, together with the nuclides in Table 2, shall be identified and reported.

e Total for parent and daughter.

Table 3

REPORTING LEVELS FOR REMP MEASUREMENTS

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg-wet)	Milk (pCi/l)	Food Products (pCi/kg-wet)
H-3	20,000[a]				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr,Nb-95	400[b]				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba,La-140	200[b]			300[b]	
Gross Beta	40	2			

a Applies to water samples utilized for human consumption only. This value is as specified in 40 CFR 141.

b Total for parent and daughter.

Table 4

SAMPLE TYPES/CLASSES

Type/Class Designation	Definition
AG	Algae
AS	Air Sample
BF	Beef Tissue
BT	Beef Thyroid
CF	Crawfish
CON	Control (CLASS)
DU	Duck
DW	Drinking Water
FG	Frog
FS	Fish Sample
HS	Honey Sample
IND	Indicator (CLASS)
LV	Garden Vegetation
MF	Milk Sample
MS	Mud & Silt (Sediment)
PH	Pheasant
PV	Pasturage
RB	Rabbit
RI	Rice
RW	Runoff Water
SG	Small Game
SL	Soil
SW	Surface Water*
TL	Direct Radiation (TLD)
WW	Well Water

Additional letter designation may be added as sample designators if additional sample types are collected for analysis.

* The portion of precipitation on the land that ultimately reaches streams is considered to be surface water.

Table 5

SECTOR DESIGNATIONS

Sector Letter	Sector Degrees and True North Compass Sector
A	348.75 <N> 11.25
B	11.25 <NNE> 33.75
C	33.75 <NE> 56.25
D	56.25 <ENE> 78.75
E	78.75 <E> 101.25
F	101.25 <ESE> 123.75
G	123.75 <SE> 146.25
H	146.25 <SSE> 168.75
J	168.75 <S> 191.25
K	191.25 <SSW> 213.75
L	213.75 <SW> 236.25
M	236.25 <WSW> 258.75
N	258.75 <W> 281.25
P	281.25 <WNW> 303.75
Q	303.75 <NW> 326.25
R	326.25 <NNW> 348.75

Table 6

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
AIR (Particulates and Iodine)			
RAS0.1C0 (RAH0)	IND	Weekly	On Site (PAP Building Carport)
RAS0.6K0 (RAD0)	IND	Weekly	Tokay Substation
RAS6.2Q0 (RAA0)	IND	Weekly	Miller Residence
RAS7.8C0 (RAF0)	IND	Weekly	Carbondale
RAS9.0E0 (RAE0)	IND	Weekly	Ione
RAS10.H0 (RAG0)	CON	Weekly	Fish Hatchery
RAS18.K0 (RAC0)	CON	Weekly	Lodi Substation
RAS23.Q0 (RAB0)	CON	Weekly	SMUD Headquarters
MILK			
RMF0.8D0 (RMFD0)	IND	Weekly	Marciel Ranch
RMF5.8P0 (RMFA0)	IND	Weekly	Mederios Dairy
RMF8.2K0 (RMFB0)	IND	Weekly	Angelo Dairy
RMF24.ØL0	CON	Weekly	DeSnayer Dairy (eff. 10/07/88)

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
RUNOFF WATER			
RRW0.3M0 (RRWC0)	IND	Biweekly	Effluent Discharge
RRW0.6M0	IND	Biweekly	S'te Boundary
SURFACE WATER			
RSW0.7N0	IND	Monthly	Water Sump
RSW1.3F0 (RSWC0)	IND	Monthly	Rancho Seco Reservoir
RSW3.7N0 (RSWB0)	CON	Monthly	Folsom South Canal Composite Sample
RSW15.5F0	IND	Monthly	Lake Pardee Reservoir (eff. 08/01/88)
RSW12.0G0 (RSWA0)	CON	Monthly	Camanche Re ervoir
RSW0.3M0	IND	Monthly	Effluent Discharge Composite Sample
DRINKING WATER			
RDW0.1G0 (RWWC0)	IND	Monthly	Rancho Seco Site Consumption (potable water)
RAIN WATER			
RRN0.6EC	IND	Seasonal	Met Station
RRN23.0Q0	CON	Seasonal	SMUD Headquarters

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
PASTURAGE			
RPV0.8D0 (RLVH0)	IND	Monthly	Marciel Ranch
RPV24.ØL0	CON	Monthly	DeSnayer Dairy (eff. 10/07/88)
RPV5.8P0 (RLVF0)	IND	Monthly	Mederios Dairy
RPV8.2K0 (RLVG0)	IND	Monthly	Angelo Dairy
WELL WATER			
RWW0.3E0 (RWWA0)	IND	Quarterly	Site Well
RWW0.8D0	CON	Quarterly	Marciel Ranch (also serves as a drinking water control sample)
RWW0.8H0 (RWWE0)	IND	Quarterly	Clay Cattle Feedlot
RWW3.7M0 (NEW)	IND	Quarterly	Silva Feed Lot
RWW2.1N0 (NEW)	IND	Quarterly	Silva Rancho Vaquero Well
RWW1.8F0 (RWWE0)	IND	Quarterly	Rancho Seco Reservoir
RWW2.1M0 (RWWB0)	IND	Quarterly	Clay Area Well

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
SEDIMENT ("MUD AND SILT")			
RMS0.3MO	IND	Quarterly	Effluent Discharge
RMS0.6MO (RMSEO)	IND	Quarterly	Site Boundary
RMS0.7NO	IND	Quarterly	Water Sump
RMS1.3FO (RMSCO)	IND	Quarterly	Rancho Seco Reservoir
RMS1.8NO	IND	Quarterly	Confluence of Clay and Hadselville Creeks
RMS2.2NO	IND	Quarterly	Hadselville Creek and Clay Station Road (eff. 1/88)
RMS14.OMO	IND	Quarterly	Laguna Creek at Twin Cities Road
RMS0.2HO	IND	Quarterly	Storm Drain Outfall 1
RMS0.2HO	IND	Quarterly	Storm Drain Outfall 2
RMS3.7NO	IND	Quarterly	Laguna Creek at Folsom South Canal
RMS5.4MO	IND	Quarterly	Laguna Creek at Laguna Road
RMS10.ONO	IND	Quarterly	Laguna Creek at McKenzie Road
RMS12.OGO	CON	Quarterly	Camanche Reservoir
FISH Include predator (e.g., bass, sunfish) and scavenger (e.g., catfish, sucker) species.			
RFS0.3MO	IND PREDATOR	Quarterly	Clay Creek near the Restricted Area Boundary

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
RFS0.3MO	IND	Quarterly SCAVENGER	Clay Creek near the Restricted Area Boundary
RFS0.6MO	IND	Quarterly PREDATOR	Clay Creek near the Site Boundary
RFS0.6MO	IND	Quarterly SCAVENGER	Clay Creek near the Site Boundary
RFS0.7NO	IND	Quarterly PREDATOR	Water sump (pond) at the Site Boundary
RFS0.7NO	IND	Quarterly SCAVENGER	Water sump (pond) at the Site Boundary
RFS1.5FO	CON	Quarterly PREDATOR	Rancho Seco Reservoir
RFS1.5FO	CON	Quarterly SCAVENGER	Rancho Seco Reservoir
RFS1.8NO	IND	Quarterly PREDATOR	Confluence of Clay and Hadselville Creeks
RFS1.8NO	IND	Quarterly SCAVENGER	Confluence of Clay and Hadselville Creeks
RFS2.2NO	IND	Quarterly PREDATOR	Hadselville Creek at Clay Station Road
RFS2.2NO	IND	Quarterly SCAVENGER	Hadselville Creek at Clay Station Road
RFS3.7NO	IND	Quarterly PREDATOR	Laguna Creek near Folsom South Canal
RFS3.7NO	IND	Quarterly SCAVENGER	Laguna Creek near Folsom South Canal
RFS5.4MO	IND	Quarterly PREDATOR	Laguna Creek at Laguna Road

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
RFS5.4MO	IND	Quarterly SCAVENGER	Laguna Creek at Laguna Road
RFS10.0MO	IND	Quarterly PREDATOR	Laguna Creek at McKenzie Road
RFS10.0MO	IND	Quarterly SCAVENGER	Laguna Creek at McKenzie Road
RFS14.0MO	IND	Quarterly PREDATOR	Laguna Creek at Twin Cities Road
RFS14.0MO	IND	Quarterly SCAVENGER	Laguna Creek at Twin Cities Road
ALGAE SAMPLES			
RAG0.3MO (RBA0)	IND	Quarterly	Effluent Discharge
RAG0.6MO	IND	Quarterly	Site Boundary
RAG0.7NO	IND	Quarterly	Water sump (pond) at the Site Boundary
RAG1.3FO	CON	Quarterly	Rancho Seco Reservoir
RAG1.8NO	IND	Quarterly	Confluence of Clay and Hadselville Creek
RAG2.2NO	IND	Quarterly	Hadselville Creek near Clay Station Road
RAG3.7NO (RBB0)	IND	Quarterly	Hadselville Creek at Folsom South Canal
RAG5.4MO	IND	Quarterly	Laguna Creek at Laguna Road
RAG10.0NO	IND	Quarterly	Laguna Creek at Twin Cities Road

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
RAG14.OMO	IND	Quarterly	Laguna Creek at McKenzie Road
SOIL			
RSLO.2JO	IND	Quarterly	Storm Drain Outfall 3
RSLO.2KO	IND	Quarterly	Storm Drain Outfall 4
RSLO.3LO	IND	Quarterly	Storm Drain Outfall 5
RSLO.2HO	IND	Quarterly	Storm Drain Outfall 6
RSLO.3MO	IND	Quarterly	Storm Drain Outfall 7
RSLO.31MO	IND	Quarterly	Storm Drain Outfall 8
RSLO.32MO	IND	Quarterly	Storm Drain Outfall 9
RSLO.5AO	IND	Quarterly	North Site Garden
RSLO.6MO	IND	Quarterly	Site Boundary
RSLO.3AO	IND	Quarterly	Storm Drain Outfall 10 (eff. 7/88)
RSLO.3QO	IND	Quarterly	Storm Drain Outfall 11 (eff. 9/88)
RS	IND	Quarterly	Storm Drain Outfall 12
F	IND	Quarterly	Silva Property
RS	CON	Quarterly	Rancho Seco Reservoir
RSL1.5NO	IND	Quarterly	Silva Property
RSL1.8NO	IND	Quarterly	Silva Property

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
CRAWFISH			
RCF0.6MO	IND	Quarterly	Clay Creek at the Site Boundary
RCF3.8PO	CON	Quarterly	Folsom South Canal
RCF0.7NO	IND	Quarterly	Water sump (pond) at the Site Boundary
RCF3.7NO	IND	Quarterly	Hadselville Creek at Folsom South Canal
RCF10.0MO	IND	Quarterly	Laguna Creek at McKenzie Road
GARDEN VEGETABLES	HARVESTED AT LEAST SEMI-ANNUALLY (SA), MONTHLY THROUGHOUT THE GROWING SEASON (MTGS)		
RLV0.5AO	IND	SA-MTGS	Broadleaf Vegetation Deposition Garden (3 broadleaf samples)
RLV0.6MO	IND	SA-MTGS	Site Boundary Vegetable Irrigation Garden (1 vegetable sample)
RLV18.KO (RLVFO)	CON	SA-MTGS	Truck Farm in the Lodi Area (3 broadleaf and 1 vegetable sample)
BEEF TISSUE			
RBFX.XNO	IND	SA	Beef Tissue From Cattle Raised West Of The Site
RBFX.XXO	CON	SA	Beef Tissue From Cattle Raised In Least Prevalent Wind Direction

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification (Former ID)	Sample Class	Collection Frequency	Location Identification
FROGS			
RFG0.6MO	IND	Quarterly	Clay Creek at the Site Boundary
RFG1.5FO	CON	Quarterly	Rancho Seco Reservoir
RFG2.2NO	IND	Quarterly	Hadselville Creek at Clay Station Road
RFG3.7NO	IND	Quarterly	Hadselville Creek at Folsom South Canal
RFG10.0MO	IND	Quarterly	Laguna Creek at McKenzie Sectors

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification	Sample Class	TLD Map #	Sample Identification	Sample Class	TLD Map #
RTL0.3RO	IND	1	RTL1.7LO	IND	21
RTL0.3CO	IND	2	RTL1.6JO	IND	22
RTL0.3NO	IND	3	RTL1.8KO	IND	23
RTL0.3LO	IND	4	RTL1.7HO	IND	24
RTL0.3HO	IND	5	RTL3.8LO	IND	25
RTL0.4FO	IND	6	RTL3.9KO	IND	26
RTL0.5CO	IND	7	RTL3.6JO	IND	27
RTL6.2QO	IND	8	RTL3.7HO	IND	28
RTL23.QO	CON	9	RTL4.2JO	IND	29
RTL18.KO	CON	10	RTL7.4MO	IND	30
RTL0.6KO	IND	11	RTL3.7NO	IND	31
RTL9.0EO	CON	12	RTL4.8PO	IND	32
RTL10.NO	IND	13	RTL3.8MO	IND	33
RTL11.MO	CON	14	RTL3.8QO	IND	34
RTL10.HO	CON	15	RTL1.9NO	IND	35
RTL2.7MO	IND	16	RTL1.6PO	IND	36
RTL8.2KO	IND	17	RTL1.9QO	IND	37
RTL7.8CO	IND	18	RTL1.6RO	IND	38
RTL1.8FO	IND	19	RTL1.5BO	IND	39
RTL1.5MO	IND	20	RTL1.5AO	IND	40

Table 6 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

Sample Identification	Sample Class	TLD Map #	Sample Identification	Sample Class	TLD Map #
RTL1.8CO	IND	41	RTL14.PO	CON	61
RTL4.4GO	IND	42	RTL11.MO	CON	62
RTL1.7FO	IND	43	RTL0.8DO	IND	63
RTL1.3FO	IND	44	RTL9.5EO	IND	64
RTL1.8EO	IND	45	RTL0.6MO	IND	65
RTL1.4DO	IND	46	RTL0.4NO	IND (eff. 7/89)	66
RTL3.0CO	IND	47	RTL0.41NO	IND (eff. 7/89)	67
RTL3.7DO	IND	48	RTL0.3PO	IND (eff. 7/89)	68
RTL3.2EO	IND	49			
RTL3.5FO	IND	50			
RTL10.EO	CON	51			
RTL19.EO	CON	52			
RTL12.GO	CON	53			
RTL11.JO	IND	54			
RTL8.OPO	IND	55			
RTL4.6QO	IND	56			
RTL7.6AO	IND	57			
RTL6.6BO	IND	58			
RTL11.RO	CON	59			
RTL11.AO	CON	60			