PROPOSED AMENDMENT 182

ATTACHMENT I

PROPOSED DEFUELED TECHNICAL SPECIFICATIONS

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D1.0 DEFINITIONS

The following terms are defined for uniform interpretation of these 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 DEFUELED MODE

The plant is in a DEFUELED MODE when the fuel has been removed from the reactor vessel and Reactor Building and there is spent fuel in the spent fuel pool. These specifications apply only after the June 1989 plant shutdown and subsequent defueling. Subsequent movement of nuclear fuel into the reactor building voids these Defueled Mode Technical Specifications.

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 defined in Specification D3, (2) it has been tested periodically in accordance with Specification D4 and has met its performance requirements, (3) the system has available its normal source of power, and (4) its required auxiliaries are capable of performing their intended function.

D1.4 INSTRUMENTATION SURVEILLANCE

D1.4.1 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.

D1.4.2 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 if there is redundant indication.

D1.4.3 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.

D1.4.4 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 SURVEILLANCE INTERVALS

The Surveillance Interval may be extended to a maximum of +25% to accommodate operations scheduling. The frequency notation specified for the performance of Surveillance Requirements shall correspond to the SURVEILLANCE INTERVALS defined in Table D1.5-1.

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 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.

TABLE D1.5-1

SURVEILLANCE INTERVALS

Frequency	Notation	Definition
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.
18 MONTHS	18M	At least once per 18 months.

D3.0 GENERAL LIMITING CONDITIONS

LIMITING CONDITION

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

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

SURVEILLANCE REQUIREMENTS

D4.0 GENERAL SURVEILLANCE REQUIREMENTS

D4.0.1 Surveillance Requirements shall be met as specified for individual Limiting Conditions unless otherwise stated in an individual Surveillance Requirement.

D4.0.2 Failure to perform a Surveillance Requirement within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval, defined by Specification D1.5, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition. The time limits of the ACTION including time requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION including time requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION including time requirements are less than 24 hours. Exceptions to these requirements are stated in the individual specifications.

D3/4.1 SPENT FUEL POOL

LIMITING CONDITION

D3.1 At least 37 feet of water shall be maintained in the spent fuel pool with the following exceptions: 1) The water level in the spent fuel pool may be less than 37 feet if the dose rate, from the irradiated core components seated in the storage racks, at the surface of the water is 2.5 mrem/hr or less and, 2) the level shall be maintained sufficient to limit the dose rate at the pool surface to 10 mrem/hr or less when core component handling activities are in progress.

APPLICABILITY: Whenever irradiated fuel assemblies or core components are in the spent fuel pool.

ACTION:

With the requirements of Limiting Condition D3.1 not satisfied, suspend movement of fuel assemblies and crane operations with loads over stored spent fuel, and restore the level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

D4.1.1 DAILY verify the level in the spent fuel pool is at least at the minimum required level.

D4.1.2 At least once every 18 MONTHS perform a calibration on the spent fuel pool level alarm switches.

Bases

When the spent fuel pool water level is maintained at a minimum of 37 feet, a minimum of 23 feet of water shielding over stored fuel assemblies is assured. This level limits radiation at the surface of the water to less than 2.5 mrem/hr. A water level of less than 37 feet is allowed as long as the dose rate at the surface of the spent fuel pool is $\leq$2.5 mrem/hr from irradiated core components seated in their storage locations in the Spent Fuel Pool.

A level limiting the dose rate at the pool surface to 10 mrem/hr provides a condition that allows the movement of fuel assemblies or core components while still limiting the exposure received by fuel handling personnel.

D3/4.2 SPENT FUEL POOL TEMPERATURE
LIMITING CONDITION

D3.2 Maintain the Spent Fuel Pool bulk coolant temperature below 140°F.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool

ACTION:

With the Spent Fuel Pool Cooling System out of service, place an alternate cooling method in service if the bulk coolant temperature is >140°F. 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.

SURVEILLANCE REQUIREMENTS

D4.2.1 DAILY verify the spent fuel pool temperature is <140°F.

Bases

This specification provides a method to ensure that the spent fuel pool bulk temperature is not maintained at or above 180°F. The use of an alternate cooling method provides alternate cooling capability to ensure 180°F is not exceeded.

D3/4.3 FUEL STORAGE BUILDING HANDLING LOAD LIMITS

LIMITING CONDITION

D3.3 Loads in excess of 1700 pounds shall be prohibited from travel over irradiated fuel assemblies in the spent fuel pool.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool

ACTION:

With the requirements of the above specification 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 crane to be used within 7 days prior to fuel handling or movement of loads over spent fuel stored in the spent fuel pool.

D4.3.2 Complete a FUNCTIONAL TEST of the Fuel Storage Building fuel handling bridge interlocks within 7 days prior to fuel handling.

Bases

The restriction on movement of loads greater than 1700 pounds over fuel assemblies in the storage pool ensures that in the event a load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) distortion of fuel in the spent fuel pool storage racks will not result in a critical array. These assumptions are consistent with the accident analyses. The rated load of the Fuel Storage Building fuel handling bridge is 2900 pounds.

D3/4.4 SPENT FUEL STORAGE RADIATION MONITOR

LIMITING CONDITION

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

APPLICABILITY: AT ALL TIMES

ACTION:

With fixed radiation monitoring equipment inoperable, fuel handling operations shall be suspended, after fuel is placed in a safe condition, until the fixed survey instrumentation is in service. Return the Fixed radiation monitor to OPERABLE status within 7 days or submit a report to the NRC, Region V documenting the condition along with a plan to return the monitor to OPERABLE status.

SURVEILLANCE REQUIREMENTS

D4.4.1 DAILY 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.

Bases

The OPERABILITY of the spent fuel storage area radiation monitor ensures that early notification is provided of excessively high radiation levels in the building.

D3/4.5 SPENT FUEL POOL WATER CHEMISTRY

LIMITING CONDITION

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 any one or more of the water chemistry limits exceeded, initiate action within 72 hours 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 of those parameters at the frequencies specified in Table D3.5-1.

Bases

The maintenance of spent fuel pool water chemistry ensures that degradation of the spent fuel assemblies and the spent fuel racks is minimized.

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 CONDITION

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, B, and C Regenerant Hold-up Tanks

b. Demineralized Reactor Coolant Storage Tank

c. Miscellaneous Water Hold-up Tank

d. Borated Water Storage Tank

e. Condensate Storage Tank

f. Outside Temporary Tanks

APPLICABILITY: AT ALL TIMES

ACTION:

With the quantity of radioactive material in any of the listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, and initiate actions to reduce the tank contents to within the limit. Reduce the tank contents to within the limit within the next 72 hours and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report pursuant to Specification D6.9.2.

SURVEILLANCE REQUIREMENTS

D4.6 The quantity of radioactive material contained in each tank listed in Specification 3.7 shall be determined to be within the specified limit by analyzing a representative sample of the tank's contents within 48 hours after radioactive materials have been added to the tank.

Bases

Restricting the quantity of radioactive material contained in the specified outdoor tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the 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 could potentially contain radioactive liquid that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to a liquid radwaste treatment system.

D3/4.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION

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 ≥0.005 microcuries of removable contamination.

APPLICABILITY: AT ALL TIMES

ACTION:

With a sealed source having removable contamination in excess of the above limits, 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 an Agreement State.

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

4.7.2 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.

D3/4.7 SEALED SOURCE CONTAMINATION (continued)

Bases

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. The use of organizations specifically authorized by the NRC or an Agreement State to perform the sealed source testing is permitted by 10 CFR 35.39.

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. Those sources which are frequently handled are required to be tested more often than those which are not. 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.

D5.0 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 minimum distance to the boundary of the exclusion area, as defined in 10 CFR 100.3(a), is 2,100 feet for Rancho Seco. The Low Population Zone is defined in 10 CFR 100.3(b).

D5.1.1 Exclusion Area

The Exclusion Area is shown in Figure D5.1-1.

D5.1.2 Low Population Zone

The Low Population Zone is shown in Figure D5.1-2.

D5.1.3 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 D5.1-3.

D5.1.4 Site Boundary For Liquid Effluent

The site boundary for Liquid Effluent for 10 CFR 20 Compliance and for meeting 10 CFR 50, Appendix I guidelines is shown in Figure D5.1-4.

FIGURE D5.1-1

EXCLUSION AREA
(2100 foot radius)

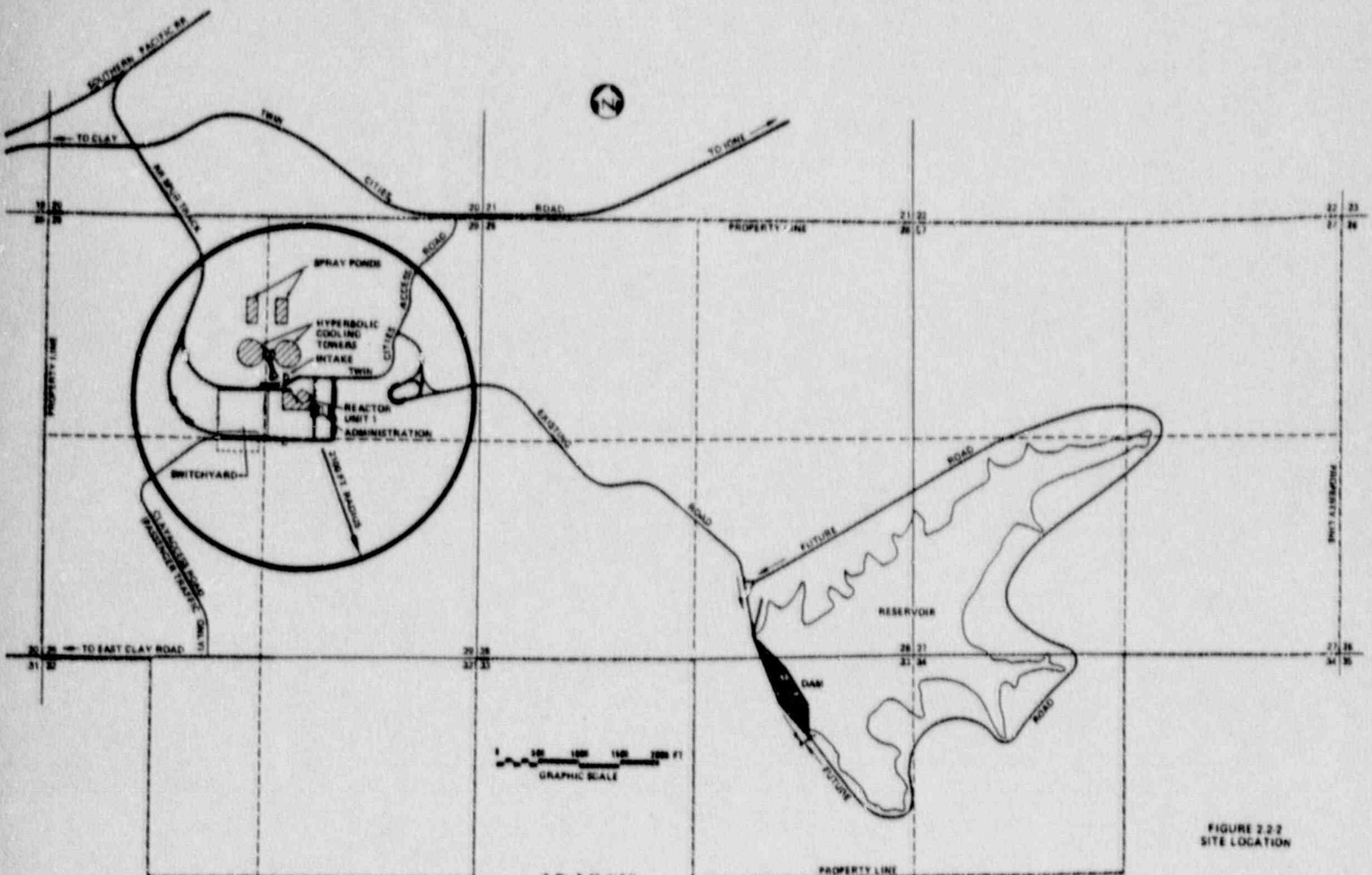


FIGURE 2.2-2
SITE LOCATION

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO UNIT 1
TECHNICAL SPECIFICATIONS

FIGURE D5.1-2

LOW POPULATION ZONE
(5 mile radius)

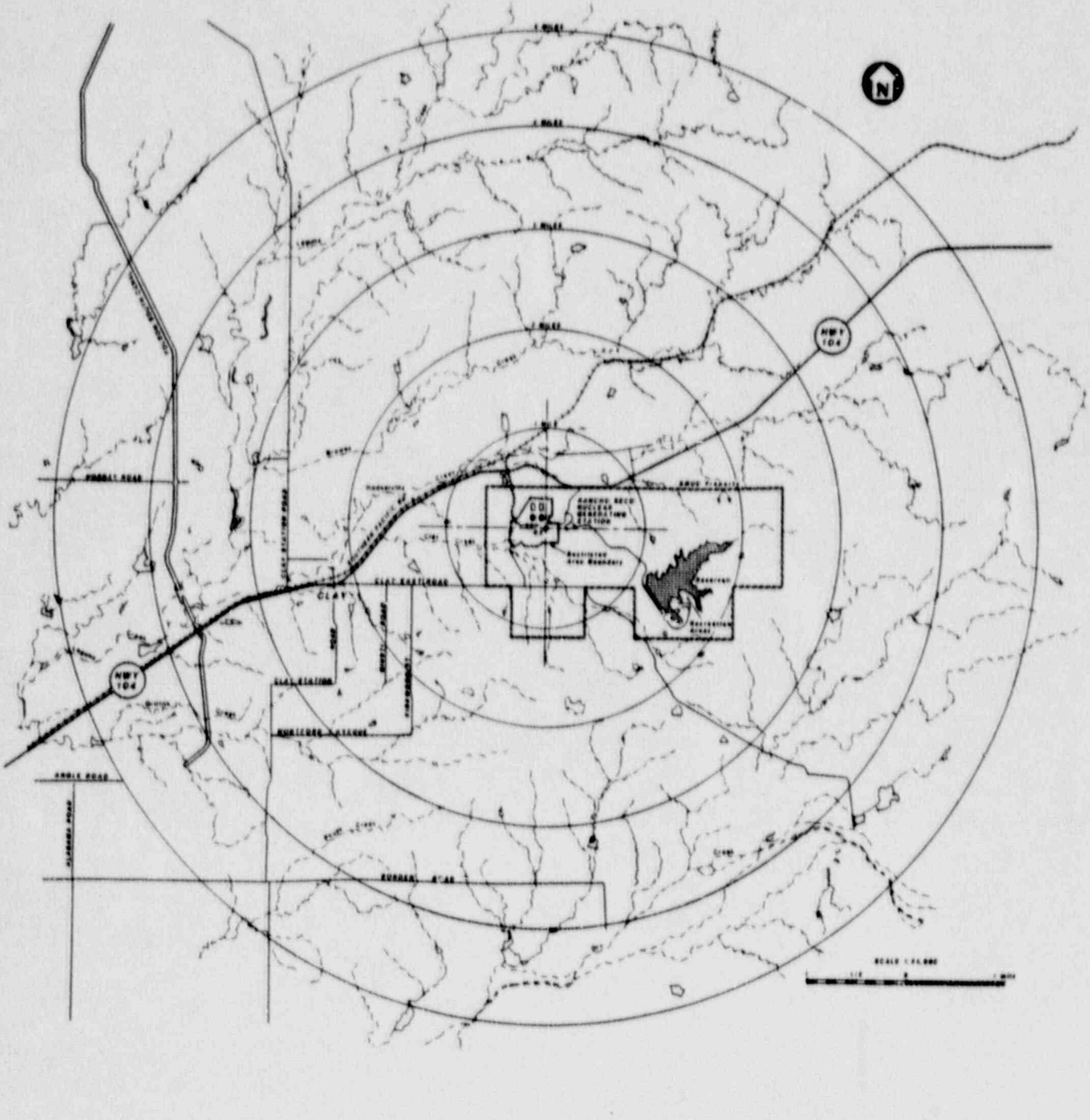


FIGURE D5.1-3

SITE BOUNDARY FOR GASEOUS EFFLUENT

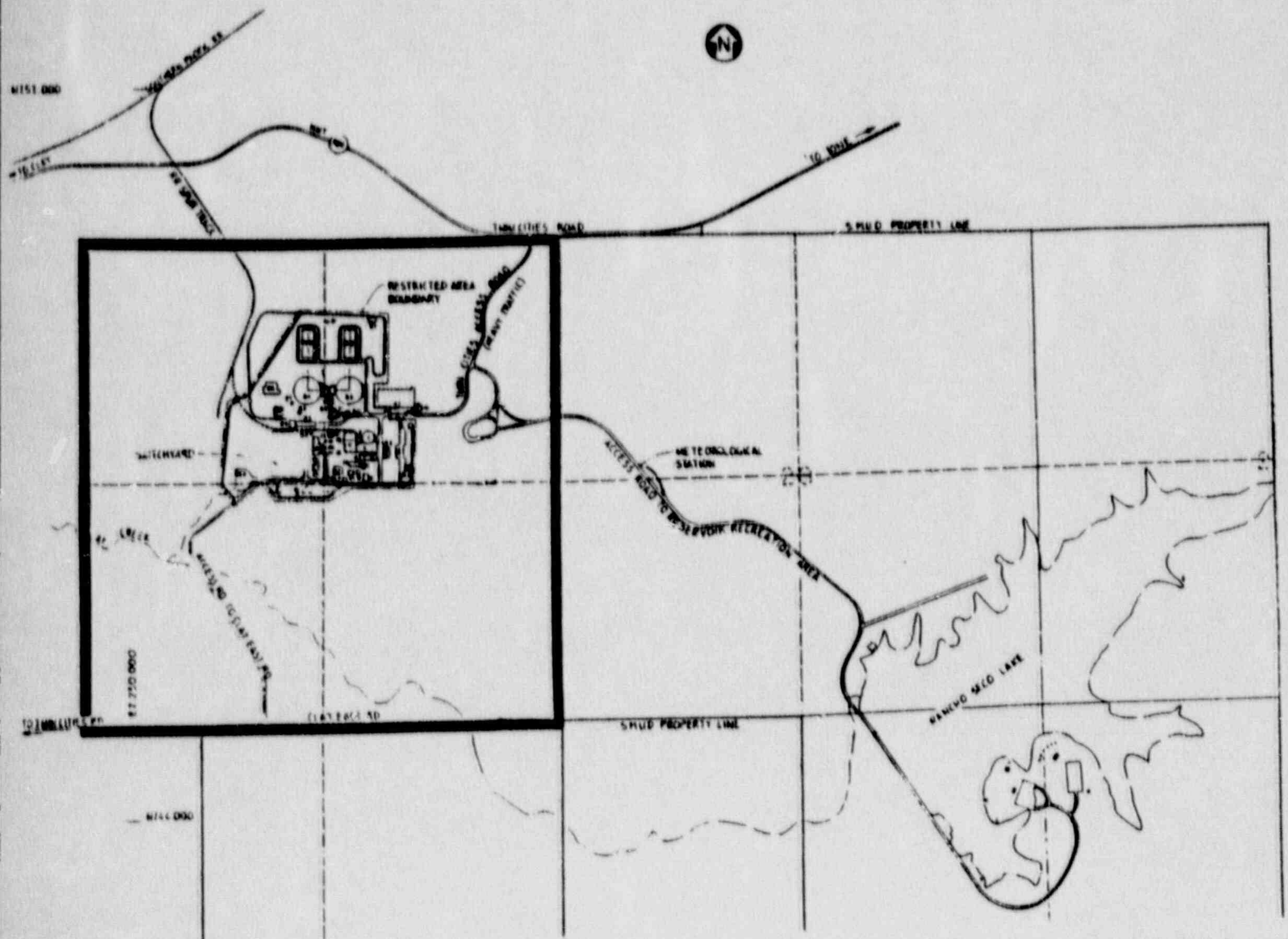
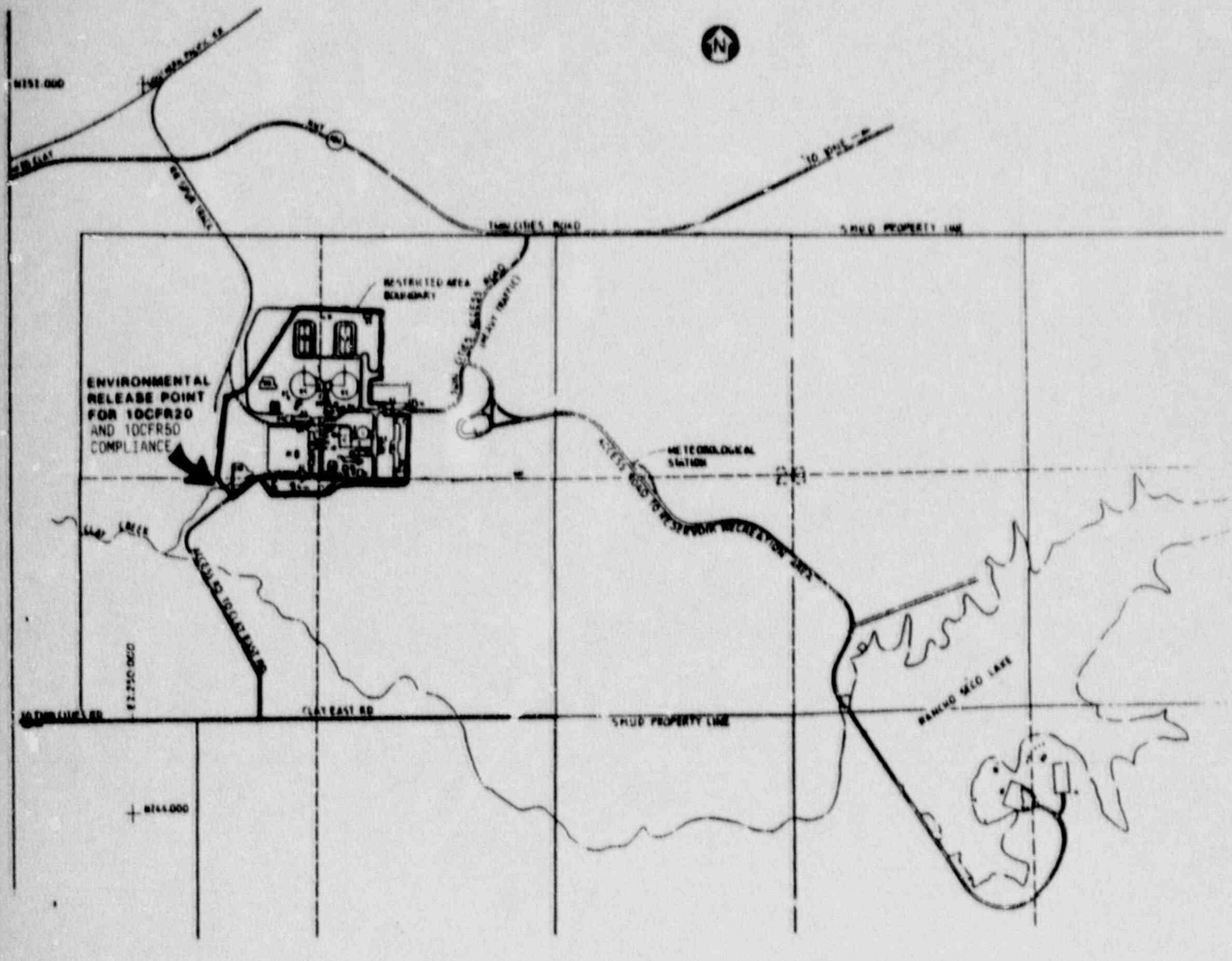


FIGURE D5.1-4

SITE 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, including four assemblies in failed fuel containers.

The pool has the capability of storing new and spent 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 and ensuring the safe storage of irradiated nuclear fuel. The Deputy AGM, Nuclear shall assume the AGM, Nuclear's responsibilities in his absence. They shall delegate in writing the succession of their responsibilities during their absences.

D6.1.2 The Shift Supervisor (or a qualified, designated individual) shall be responsible for the control room command function.

D6.2 ORGANIZATION

D6.2.1 An organization shall be established for facility operation. The organization shall include the positions responsible for activities affecting the safe storage of irradiated nuclear fuel.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels, including all 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 Updated Safety Analysis Report (USAR).

b. The AGM, Nuclear shall have responsibility for and shall take measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The individuals who train the operating staff 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 scheduler pressures.

D6.2.2 The facility organization shall be as shown in the USAR and each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table D6.2-1. In addition, the Shift Operations Superintendent shall hold a Senior Reactor Operator license in accordance with 10 CFR 55.

TABLE D6.2-1

MINIMUM SHIFT CREW REQUIREMENTS*

Position	Defueled
Shift Supervisor or Control Room Operator	1 - L **
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.

L - NRC Licensed Operator _or_ NRC Senior Licensed Operator

** - In addition to the on-shift NRC Licensed Operator, an individual possessing an NRC Senior Operator License shall be on-call pursuant to the requirements of 10 CFR 50.54(m)(1).

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 Manager, Radiation Protection who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

D6.4 TRAINING

A retraining and replacement training program for the Licensed Operators shall be maintained under the direction of the AGM, Nuclear and shall be conducted in accordance with the NRC approved training program.

D6.5 REVIEW AND AUDIT

D6.5.1 Plant Review Committee (PRC)

D6.5.1.1 The PRC shall function to advise the Deputy AGM, Nuclear on matters related to nuclear safety.

D6.5.1.2 The PRC shall be composed of a Chairman and a minimum of six members.

D6.5.1.3 Membership of the PRC shall be as defined below:

a. A list of members made up of personnel filling positions in the Nuclear Organization and others who meet or exceed the minimum qualifications of ANSI N18.1-1971, Section 4.2 or 4.4, shall be developed and approved by the Deputy AGM, Nuclear.

b. A list of alternate members of the PRC shall be developed and approved by the Deputy AGM, Nuclear.

c. The Chairman of the PRC shall be appointed by the Deputy AGM, Nuclear. When a meeting is scheduled or an emergency meeting is necessary, and the PRC Chairman will be absent, an alternate Chairman may be designated by the PRC Chairman or the Deputy AGM, Nuclear from among the PRC members in Specification D6.5.1.2.

D6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chairman, or as directed by the Deputy AGM, Nuclear.

D6.5.1.5 A Quorum of the PRC shall consist of a majority of the members in Specification D6.5.1.2 including the Chairman or alternate Chairman. No more than two alternates shall participate in PRC activities at any one time for the purpose of establishing a quorum.

ADMINISTRATIVE CONTROLS

D6.5.1.6 The PRC shall be responsible for review of:

a. The required safety evaluation of: (1) all procedures and programs required by Specification D6.8 and changes thereto, and (2) any other proposed procedures and programs or changes thereto which are as determined by the Deputy AGM, Nuclear to affect nuclear safety.

b. The safety evaluations of proposed tests and experiments that affect nuclear safety.

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

d. Safety evaluations of proposed changes or modifications to plant systems or equipment that affect nuclear safety. Items which are determined by a qualified reviewer as not involving an unreviewed safety question, a change of Technical Specifications, or a change in a licensing basis document need not be reviewed by the PRC.

e. Investigations of all violations of the Technical Specifications to determine adequacy of corrective action and to detect any degrading trend.

f. Facility operations to detect potential safety hazards.

g. Events requiring a Licensee Event Report as defined by 10 CFR 50.73 and NUREG-1022 to determine adequacy of corrective action and to detect any degrading trend. (See Specification D6.9.4)

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

i. The Plant Security Plan and changes thereto.

j. The Emergency Plan and changes thereto.

k. The Fire Protection Plan and changes thereto.

l. Changes to the PROCESS CONTROL PROGRAM (PCP), the OFFSITE DOSE CALCULATION MANUAL (ODCM) and the RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) MANUAL. (See Specifications D6.13 and D6.14)

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

n. Any accidental, unplanned, or uncontrolled release of radioactive material to the environs, including the preparation and forwarding of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence, and the forwarding of these reports to the Deputy AGM, Nuclear and to the MSRC.

D6.5.1.7 The PRC shall:

a. Recommend in writing to the Deputy AGM, Nuclear approval or disapproval of items considered under Specifications D6.5.1.6.

b. Render determinations in writing with regard to whether or not each item considered under Specifications D6.5.1.6a through 6e, and 6m above constitutes an unreviewed safety question.

c. Provide immediate written notification to the MSRC Chairman of disagreement between the PRC and the Deputy AGM, Nuclear; however, the Deputy AGM, Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification D6.5.1.1.

d. Form subcommittees to screen reviews designated in Section D6.5.1.6.

D6.5.1.8 Minutes of each PRC meeting, including appropriate documentation of reviews encompassed by Specification D6.5.1.6(e) and (g), shall be prepared, approved, and forwarded to the Deputy AGM, Nuclear and to the MSRC Chairman within 14 days following each meeting.

D6.5.2 MANAGEMENT SAFETY REVIEW COMMITTEE (MSRC)

D6.5.2.1 The MSRC shall provide independent review and audit on matters related to nuclear safety and protection of the health and safety of the public.

D6.5.2.2 The MSRC shall be composed of a Chairman and a minimum of six members. THE MSRC membership shall be made up of manager levels or above in the Nuclear Organization or of other personnel meeting or exceeding the minimum qualifications of ANSI/ANS 3.1-1981, Section 4.7.2.

D6.5.2.3 The Chairman of the MSRC shall be appointed by the AGM, Nuclear. A current list of members and alternates in Specification D6.5.2.2 shall be developed and approved by the AGM, Nuclear. In the absence of the Chairman, an alternate Chairman may be designated by the MSRC Chairman of the AGM, Nuclear. The alternate Chairman shall be selected from among the MSRC members in Specification D6.5.2.2.

D6.5.2.4 The MSRC shall meet at least once per calendar quarter and as convened by the MSRC Chairman or as directed by the AGM, Nuclear.

D6.5.2.5 A quorum of MSRC shall consist of a majority of the members in Specification D6.5.2.2, including the Chairman or alternate Chairman. No more than two alternates shall participate in MSRC activities at any one time for the purpose of establishing a quorum.

ADMINISTRATIVE CONTROLS

D6.5.2.6 The MSRC shall be responsible for review of:

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

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

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

d. Proposed changes in Technical Specifications or the Operating License.

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

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

g. LICENSEE EVENT REPORTS as defined by 10 CFR 50.73 and NUREG-1022.

h. Any indication of an identified significant deficiency in the design or operation of safety-related structures, systems, or components.

i. Reports and meeting minutes of the PRC.

j. Any facility activity brought to the attention of the MSRC by the District's executive management which may be indicative of conditions adverse to nuclear safety.

D6.5.2.7 The MSRC shall:

a. Report to and advise the AGM, Nuclear of those areas of responsibility specified in Specification D6.5.2.6.

b. Recommend to the AGM, Nuclear other areas of facility operation for which the MSRC has determined the need for additional auditing per Specification D6.5.4(g).

c. Advise the AGM, Nuclear of the need for independent auditing of facility operations.

D6.5.2.8 Minutes of each MSRC meeting and separate documentation of reviews encompassed by Specifications D6.5.2.6(d),(e),(f),(g) and (h) shall be prepared, approved, and forwarded to the AGM, Nuclear within 14 days following each meeting.

D6.5.3 Technical Review and Control

Activities which affect nuclear safety shall be conducted as follows:

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

   The AGM, Nuclear will approve plant administrative procedures, Radioactive Effluent Controls Program implementing procedures, REMP implementing 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 procedures can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. The change shall be documented, reviewed and approved by the procedures approval authority within 14 days of implementation.

ADMINISTRATIVE CONTROLS

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 the Deputy AGM, Nuclear.

c. Proposed tests and experiments which affect plant 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 the Deputy AGM, Nuclear.

d. Individuals responsible for reviews performed in accordance with Specification D6.5.2a, D6.5.2b, and D6.5.2c shall meet or exceed the qualification requirements of Section 4.4 of ANSI 18.1-1971. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary. A list of qualified reviewers for the independent reviews described in D6.5.2a, b, and c above shall be established by the AGM, Nuclear. The personnel performing the cross-disciplinary review need not be qualified reviewers, but a qualified reviewer shall review each determination.

e. Events reportable pursuant to Technical Specification 6.9 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 PRC and forwarded to the AGM, Nuclear.

D6.5.4 Audits

Audits of facility activities shall be performed under the cognizance of the Manager, Quality and Safety. 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 District's 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 at least once per 12 months.

d. The performance of activities required by the Quality Assurance Program at least once per 2 years.

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

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

g. Any other area of facility operation 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 utilizing either qualified offsite licensee personnel or an outside fire protection firm.

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

k. The radiological environmental monitoring program and the results thereof at least once per 24 months.

l. The ODCM and implementing procedures at least one per 24 months.

m. The PCP and implementing procedures for processing and packaging of radioactive wastes from liquid systems at least once per 24 months.

n. The performance of activities required by Rancho Seco procedures and programs for effluent control and environmental monitoring.

ADMINISTRATIVE CONTROLS

D6.5.4 Audits (Continued)

Audit reports of reviews encompassed by Specification D6.5.3 shall be forwarded to the AGM, Nuclear and to 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 D6.9.4, pursuant to the requirements of 10 CFR 50.73, and

b. Each Licensee Event Report shall be reviewed by the PRC and the results of this review submitted to the MSRC. Each Licensee Event Report shall be reviewed and approved by the AGM, Nuclear, or designee.

ADMINISTRATIVE CONTROLS

D6.7 - NOT USED -

D6.8 PROCEDURES AND PROGRAMS

D6.8.1 Written procedures 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. Fuel Storage, Decommissioning, and Decontamination
c. Surveillance and test activities of safety-related fuel storage equipment
d. Security Plan
e. Emergency Plan
f. Fire Protection Plan
g. PCP
h. ODCM
i. REMP MANUAL
j. Quality Assurance Program for Effluent Control and Environmental Monitoring using the guidance of Regulatory Guide 4.15, Revision 1, February 1979

D6.8.2 Each procedure 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 achievable. 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,

ADMINISTRATIVE CONTROLS

D6.8 PROCEDURES AND PROGRAMS (Continued)

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 are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 8 1/3%(1/12) of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 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 Iodine-131, Iodine-133, 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,

ADMINISTRATIVE CONTROLS

D6.8 PROCEDURES AND PROGRAMS (Continued)

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 a 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 and D6.14.

D6.9 REPORTING REQUIREMENTS

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

D6.9.1 Annual Radiological Reports

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

D6.9.1.1 Annual Occupational Radiation Exposure Report

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

D6.9.1.2 Annual Exposure Report

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

D6.9.1.3 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the activities 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 content of this report shall be consistent with the objectives outlined in (1) the REMP MANUAL and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ADMINISTRATIVE CONTROLS

D6.9.2 Semiannual Radioactive Effluent Release Report

The Semiannual Radioactive Effluent Release Report covering the activities of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The content of this report shall be consistent with the objectives outlined in (1) the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and (3) consistent with Section IV.B.1 of Appendix I to 10 CFR Part 50.

D6.9.3 ANNUAL REPORT

A routine report consisting of shutdown statistics, a narrative summary of shutdown experience, major safety-related maintenance, 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.

D6.9.4 LICENSEE EVENT REPORT

The types of events listed in 10 CFR 50.73 shall be the subject of Licensee Event Reports, submitted to the U.S. Nuclear Regulatory Commission, 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 to prevent recurrence. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, a Licensee Event Report shall be completed and reference shall be made to the original report date, pursuant to the requirements of 10 CFR 50.73.

D6.9.5 Environmental Reports

a. When a change to the plant design or to the plant operation 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.

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 required changes are submitted to the concerned agency 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.

D6.10 RECORD RETENTION

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

a. Records and logs of Facility activities.

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 changes made to Operating Procedures.

f. Records of radioactive shipments.

g. Records of sealed source leak tests and results.

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

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

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Updated Safety Analysis Report.

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

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

D6.10 RECORD RETENTION (Continued)

j. Records of meetings of the PRC and the MSRC.

k. Records for the Radiological Environmental Monitoring Program.

l. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and Maintenance records.

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

n. Records of Decommissioning activities performed.

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

In lieu of the "control device" or "alarm signal" required by of 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, and 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 D6.12.1(a) above, and, in addition, locked doors shall be provided to prevent unauthorized entry into such area, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty. Certain areas within the controlled area may use conspicuous visible or audible signals such that an individual is made aware of the presence of the High Radiation Area, in lieu of locked doors.

ADMINISTRATIVE CONTROLS

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 shall:

a. Be documented and records of reviews performed shall be retained as required by Specification 6.10.21. 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. Become effective after review and acceptance by the PRC 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 Changes to the ODCM or REMP MANUAL shall be made as follows:

a. Licensee-initiated changes shall be documented and records of reviews performed shall be retained as required by Specification D6.10.21. 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 Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 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 PRC 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.

PROPOSED AMENDMENT 182

ATTACHMENT II

SAFETY ANALYSIS AND NO SIGNIFICANT HAZARDS CONSIDERATION

Safety Analysis Log No. 1091
Proposed Amendment No. 182

Description of Change

This Proposed Amendment (PA) No. 182 creates "Appendix C" to Facility Operating License No. DPR-54. The "Appendix C," or Defueled Technical Specifications (DTSs) will be followed in lieu of the Appendix A and B Technical Specifications for the remainder of the extended outage that began in June 1989. The DTSs are based on the conditions present at Rancho Seco on January 1, 1990 and thereafter. Movement of fuel into the Reactor Building will require that the plant revert back to the Appendix A and B Technical Specification requirements. Deletion of the Appendix B Technical Specifications was previously justified in Proposed Amendment 102, Revision 1. The Administrative Controls section of the DTSs provide sufficient requirements to allow the deletion of the Appendix B Technical Specification in the Defueled Mode.

Reason For Change

The Rancho Seco DTSs are based on the plant conditions that exist as a result of the extended outage that the plant entered in June 1989. Fuel has been off-loaded from the reactor vessel and stored in the spent fuel pool (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 DTSs provide a level of protection needed to deal with credible accidents in the defueled condition, provide safe long term storage of the nuclear fuel, and provide the necessary level of protection of health and safety of the public.

The creation of the DTSs provides a clear and concise set of Specifications applicable in the Defueled Mode. These Specifications eliminate those requirements not needed in the Defueled Mode.

The DTSs are submitted as an addition to the current Technical Specifications and are intended to be Appendix C to Facility Operating License No. DPR-54. Upon NRC approval these DTSs will be followed in lieu of the Appendix A and B Technical Specifications for the remainder of the current extended outage. The current outage will end with the movement of fuel into the

Reactor Building.

Evaluation and Basis For Safety Analysis

The primary difference between the proposed DTSs, and the current Rancho Seco Technical Specifications is the range of postulated accidents that must be dealt with and the potential radiological consequences of abnormal plant conditions in the Defueled Mode. Based on a review of these conditions and credible accidents, the DTSs were developed to provide the necessary level of protection.

The Updated Safety Analysis Report (USAR), Chapter 14, contains two accidents or conditions that are applicable in the Defueled Mode:

1. Fuel Handling Accident

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

These are the two accidents or conditions that are considered credible in a defueled condition and were thus the two considered in the analysis below.

Several Generic Letters, as listed below, were incorporated in the DTSs as follows:

a. 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 submittal for Generic Letter 88-12, REMOVAL OF FIRE PROTECTION REQUIREMENTS FROM TECHNICAL SPECIFICATIONS, is included in this submittal by omission of the appropriate Technical Specifications as delineated in Proposed Amendment 180.

The primary concerns in the Defueled Mode are protection of the spent fuel in the SFP, and control of radiological releases to the environment. The controls required to protect the spent fuel are predicated primarily on the level of decay heat in the SFP. The decay heat load for the SFP in the current extended shutdown was calculated using the methodology described in ANSI/ANS 5.1-

1979 and BTP ASB9-2. The decay heat load in the SFP, as a function of time, is shown in Table 1. Table 1 also provides, as a function of calendar date, the time required for the SFP to reach 212 degrees Fahrenheit, and the time required to boil 6.75 feet of water from the SFP. The 6.75 feet was chosen since the SFP must be maintained at 30 feet 3 inches to limit the dose rate at the SFP surface to 2.5mr/hr when a full core with 30 days decay time is off-loaded, and the initial SFP level is assumed to be 37 feet. This calculation is very conservative since the reactor was not operated at 100% power as assumed in the dose rate calculation, and the decay time of 207 days is significantly higher than the 30 days assumed. In addition, there is no safety implication for a significantly lower SFP level so long as personnel exposure is monitored and maintained as low as reasonably achievable.

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 10% conservatism factor.

3. There is no heat transfer loss from the SFP due to evaporation or ambient losses, except for that which occurs when the SFP reaches the boiling point.

As can be seen from Table 1, with a loss of all Spent Fuel Pool Cooling (SFC) the minimum time required to reach boiling in the SFP is 74.4 hours on January 3, 1990, and increases to 96.12 hours by June 7, 1990. Thus, a minimum of 3 days, and in the near future 4 days, will be available to restore SFC, prior to the occurence of boiling, on a loss of cooling for the SFP. In addition, if the extra time available through boil-off of 6.75 feet of the 37 foot normal water depth is taken into account, a minimum of 9.18 days on January 3, 1990, and 11.86 days by June 7, 1990 will be available for corrective actions to be implemented. There are, however, no negative effects on the fuel associated with SFP boiling since the fuel is designed to operate with significantly higher coolant temperatures than 212 degrees Fahrenheit. The primary concern with boiling in the SFP is the effect of elevated temperatures on the structural support of the SFP. The thermal stresses on the SFP, however, have been analyzed and found acceptable up to a SFP temperature of 212 degrees Fahrenheit. Therefore, a 212 degree SFP temperature does not impose a significant safety hazard.

During normal plant operations and post accident it is imperative that electrical power be available immediately to support equipment needed to mitigate the consequences of an accident. In the Defueled Mode, however, there is a significant amount of time available, at least 74.4 hours as of January 3, 1990, to take corrective action without significant safety consequence in the event of a LOOP. Rancho Seco has six off-site power transmission lines, and has the capability to receive power directly from the District's or PG&E's hydroelectric units in less than eight hours in the event of a LOOP. An evaluation of the off-site electrical grid for Rancho Seco was performed to satisfy 10CFR50.63, Station Blackout, and 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 is available in the time period necessary to support SFP cooling equipment.

Criticality control in the SFP is achieved through the use of high density fuel storage racks. These racks are designed to hold 1,080 fuel assemblies of 4.0 weight percent enrichment with unborated water while still maintaining K_{eff} less than 0.95. The 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 (remaining U-235 plus fissile Pu present) enrichment for any assembly in the SFP is 2.573 weight percent, thus providing an even greater shutdown margin. Also, the control assemblies are stored in the spent fuel which provides additional neutron absorption material. Since movement of fuel into the reactor is not allowed under the Defueled Technical Specifications (automatic reversion to the Appendix A Technical Specifications when fuel is moved into the reactor building), and no more than two fuel assemblies at a time can be moved within the SFP, there is no need for boron in the SFP to maintain adequate shutdown margin.

The SFP storage rack moderation capability is assured through a coupon sampling program that periodically removes Boraflex material samples from the SFP for testing to verify rack performance. To date, the removed coupons have provided verification of the Boraflex integrity and rack performance as designed. In addition, the DTSs contain new requirements to monitor the SFP water for chloride and fluoride levels to help 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 intact and perform their intended function.

The spent fuel at Rancho Seco has undergone substantial decay

(207 days as of January 1, 1990) and as such the source term for a credible accident is small. The spent fuel handling accident, the worst case accident in the defueled condition, will not result in a release that exceeds either the 10CFR100 accident dose limits or the 10CFR50, Appendix I annual dose objectives. This conclusion is supported by a District calculation and the safety evaluation that accompanied current Technical Specification 3.13.2 which requires only 30 days decay time prior to fuel movement when the Auxiliary and Spent Fuel Building filter system is not Operable. The District expects only limited fuel handling activities prior to permanent removal of the spent fuel from Rancho Seco.

The following is a line by line discussion of the current Technical Specifications and their disposition in relationship to the DTSs.

Technical Specification Number

1.1 Rated power is not used in the DTSs and is not included.

1.2 Definitions 1.2.1 through 1.2.12, and 1.2.14 are related to plant operations or non defueled activities, are not used in the DTSs, and are not included. Specification 1.2.13 is included as DTS D1.1. A definicion for the Defueled Mode is included as DTS D1.2.

1.3 The definition for OPERABLE has been modified to remove both the requirement for an emergency power source, and the statement that a system or component may be considered operable with only its normal or emergency power source operable as long as its redundant system or component has OPERABLE normal and emergency power sources. This modified definition is included as DTS D1.3

1.4 These Definitions apply to systems such as SFAS and RPS that are not in service in the Defueled Mode. These terms are not used in the DTSs and are not included in the Definition section.

1.5 Those Definitions that are required to describe terms used in the DTSs, i.e. Channel Test, Instrument Channel Check, and Instrument Channel Calibration, are included as Definitions D1.4.1, D1.4.2, and D1.4.3, respectively. The remaining Definitions are not used and are not included.

1.6 This Definition describes core reactivity conditions that are not applicable with the reactor defueled and are therefore not included.

1.7 This Specification defines Containment Integrity which is not applicable in the Defueled Mode. Therefore, this definition is not included in the DTSs.

1.8 The description for Licensee Event Reports is included in DTSs D6.9.4.

1.9 The applicable surveillance intervals are included as DTSs D1.5.

1.10 The term SAFETY is not used in the DTSs in the context that it is used in the current Technical Specifications and is therefore not included.

1.11 This Specification defines Fire Protection terms that were removed in Proposed Amendment 180 and are therefore not included.

1.12 This Definition, Staggered Test Basis, is not used in the DTSs and is not included.

1.13 The Definition for the Process Control Program (PCP) is included as Definition D1.6 and has been modified per the guidelines of Generic Letter 89-01.

1.14 This Definition, Solidification, has been replaced by the term "Processed" and relocated to the PCP per the guidelines of Generic Letter 89-01. This new term will allow the use of new technics that meet NRC and State requirements for disposal of wastes.

1.15 The Definition for the Offsite Dose Calculation Manual (ODCM) has been modified to meet the guidelines of Generic Letter 89-01 and included as DTS D1.7.

1.16 This Restricted Area is described in DTSs Section D5, is not used within the body of the DTSs, and is therefore not included in the Definition section.

1.17 The Site Boundary is defined in DTSs Section D5, is not used within the body of the DTSs, and is therefore not included in the Definition section.

1.18 This Definition, Dose Equivalent I-131, is relocated to the ODCM per Generic Letter 89-01.

1.19 This Definition, Member(s) Of The Public, is included as Definition D1.9.

1.20 This Definition, Dewatering, has been replaced by the term "Processed" and relocated to the PCP per Generic Letter 89-01. This new term will allow the use of new technics that meet NRC and State requirements for disposal of wastes.

1.21 This Definition, Maximum Exposed (hypothetical) Individual, is relocated to the ODCM per Generic Letter 89-01.

1.22 The definition of the Radiological Environmental Monitoring Program (REMP) Manual has been modified per Generic Letter 89-01 and is included as DTS D1.10.

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, was relocated to the ODCM per Generic Letter 89-01.

1.26 This Definition, Venting, was relocated to the ODCM per Generic Letter 89-01.

1.27 This Definition, E-BAR, is for radionuclide concentrations in the RCS based on beta and gamma activity and is thus not applicable in the Defueled Mode.

2 Section 2 of the Specifications provides limits for core and RCS operations. These are not applicable with the reactor defueled. Therefore, this Section is not used.

3.0.1 This Specification has been modified to reflect the presence of only one Mode in the Appendix C DTSs.

3.0.2 Incorporated with a change from "Limiting Condition for Operation (LCO)" to "Limiting Condition (LC)" since the plant is not permitted to operate in the Defueled Mode.

3.0.3 This Specification provides time periods by which Mode changes must be made when a Limiting Condition for Operation is not met. This Specification is not applicable since the plant is in only one Mode.

3.0.4 This Specification provides restrictions for entry into an Operational Mode. This Specification is not applicable since the plant is in only one Mode.

3.1 Specifications 3.1.1.1 through 3.1.9.2 are not required to be met in the Cold Shutdown/Refueling Shutdown/Refueling Operations Modes. These Specifications are not applicable since they provide for protection of the RCS and the reactor core which is not a safety concern with the reactor defueled and the RCS depressurized.

3.2 Specifications 3.2.1.1 through 3.2.2.6 provide for core criticality control and Low Temperature Overpressure Protection of the RCS. Since the heat/pressurization source and the criticality source (the core) have been removed, there is no need to provide this protection. Should an inadvertent pressurization occur, the effect on the health and safety of the public is not significant with fuel removed from the core.

3.3 Specifications 3.3.1 through 3.3.2 provide for Emergency Core Cooling, Reactor Building Emergency Cooling, and Reactor Building Spray systems and are not applicable in the Cold Shutdown/Refueling Shutdown/Refueling Operations Modes. Since the reactor is defueled, there are no postulated accidents that require these systems to be operational.

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 in the Cold Shutdown/Refueling Shutdown/Refueling Operations Modes since the RCS is not at a temperature where the steam generators are in service. Therefore, these Specifications are not included in the DTSs.

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 Process Instrumentation to be Operable for startup of the plant. These Specifications are not applicable in the Cold Shutdown/Refueling Shutdown/Refueling Operations Modes (except Table 3.5.1-1, Process Instrumentation Item 9). Item 9 provides for isolation of the reactor building purge system on high radiation levels through the system. Since there are no postulated accidents in the reactor building with the reactor defueled, and the aiborne activity lavels in the building are extremely low, the isolation system is not required. Therefore, these Specifications are not included in the DTSs.

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 the Cold Shutdown/Refueling Shutdown/Refueling Operations Modes. Since the reactor is defueled, these Specifications are not included in the DTSs.

3.5.3 Specification 3.5.3 provides setpoints for the SFAS system and is not applicable in the Cold Shutdown/Refueling Shutdown/Refueling Operations Modes. Since the accidents mitigated by SFAS system actuation are not credible in the defueled condition, these Specifications are not included in the DTSs.

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 DTSs 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 plant 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 DTSs. The postulated accidents these instruments are required to monitor are not credible in the defueled Mode.

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 DTSs since the steam generators that the EFIC system supports are not required in the Defueled Mode.

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. Thus, this Specification is not included in the DTSs.

3.6 Specifications 3.6.1 through 3.6.8 provide containment integrity requirements to ensure that releases to the environment post accident are maintained as low reasonably achievable. Since there are no postulated accidents inside containment in the Defueled Mode, these Specifications are not included in the DTSs.

3.7 Specifications 3.7.1 through 3.7.4 provide Operability requirements for the normal and emergency power sources. As previously addressed in this submittal, a significant amount of time is available to take corrective action to restore offsite power in the unlikely event of a LOOP. Therefore, these Specifications are not required to ensure power is available in a timely manner to support needed equipment, and they are not included in the DTSs.

3.8 Specification 3.8 provides requirements for fuel handling operations. The Specification 3.8.1 radiation monitor associated with the SFP is included as DTS D3/4.4. The radiation monitors associated with the reactor building are not included since no fuel handling in the reactor building is allowed under the DTSs.

Specifications 3.8.2 through 3.8.11, and 3.8.13 are associated with fuel movement in the Reactor Building, are not applicable in the Defueled Mode, and are not included in the DTSs.

The requirements of Specification 3.8.12, control of heavy loads over the SFP, are included in DTS D3/4.3.

3.9 Several systems other than SFC are available to provide cooling capability for the SFP. These include DHR, and Reactor Building Spray on recirculation to the Borated Water Storage Tank (BWST). In addition, the District is considering the installation of a new backup SFP cooling system that would use a single loop cooling system with a water to air radiator. Any of these cooling systems will provide effective backup SFP cooling. There is no need to restrict the options of plant Operations personnel to only one method of backup cooling as is done in Technical Specification 3.9.1. Therefore, Technical Specification 3.9.1 is not included in the DTSs, although a temperature limit for the SFP is included in DTS D3.2.

Specifications 3.9.2 and 3.9.3 provide restrictions on the use of DHR as a SFP cooling source when it is required for RCS protection. In the Defueled Mode the DHR system is not required for protection of the reactor. Therefore, these Specifications are not included in the DTSs.

Specification 3.9.4, SFP temperature limit, is modified to remove the reactor operation limits, and is included in DTS D3/4.2.

Specification 3.9.5, SFP water level is included in DTS D3/4.1.

3.10 Specification 3.10 provides a limit for the amount of Iodine-131 in the secondary side of the steam generators. Since both the primary and secondary side water of the steam generators are currently well below the required Iodine-131 level, and there is no new production of Iodine-131 in the Defueled Mode, this Specification is not included in the DTSs.

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. Therefore, this Specification is not included in the DTSs.

3.12 Specification 3.12 provides testing requirements for snubbers on safety related systems. There is only one safety related snubber on the SFC system. This snubber will be maintained under a testing and inspection program consistent with the current Technical Specification required program. With only one snubber involved, there is no justification for a detailed

Technical Specification. The maintenance and testing will be performed to ensure continued acceptable performance.

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. Containment integrity is not required in the defueled condition. The CR/TSC system is not required since the source term is low enough, due to fission gas decay, that a fuel handling accident will not result in an exposure to Control Room personnel in excess of 10CFR100 allowable levels. In addition, the District Administratively maintains chlorine levels on-site at or below 100 pounds to restrict Operator exposure to chlorine in the event of a spill.

The Reactor Building Purge filter system is not required since there is no fuel in the Reactor Building, airborne radioactivity levels in the building are low, and purges of the building are controlled by the ODCM requirements.

Specification 3.13.2 provides the requirements for the Auxiliary and Spent Fuel Building filter system. This system is required only when the plant is in operation or when fuel handling is in progress and the fuel has decayed for less than 30 continuous days. Since the fuel has decayed for greater than 30 days this system is not required for the DTSs.

3.14 Specifications 3.14.1 through 3.14.6.2 are Fire Protection Technical Specifications that were justified for removal pursuant to Generic Letter 88-12 in the District's submittal of Proposed Amendment 180.

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
3.18.1 Gaseous Effluents Dose Rate
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
3.25 Fuel Cycle Dose
3.26 Interlaboratory Comparison Program

The Specifications listed above (3.15 through 3.26) have been removed from the Technical Specifications pursuant to the requirements of Generic Letter 89-01 and placed in the Off-site Dose Calculation Manual (ODCM). The updated ODCM is included with this submittal for NRC review.

Specification 3.18.1 was modified in the ODCM to use the Site Boundary for Gaseous Effluents, Defueled Technical Specification Figure D5.4, in lieu of the Exclusion Area currently used for "Dose Rate" calculations. This change is consistent with current "Dose" calculations and NUREG-0133, Preparation Of Radiological Effluent Technical Specifications For Nuclear Power Plants.

3.17.3 This Specification provides radioactive material content limits for outdoor liquid holdup tanks and is included as DTS D3.6.

3.18.5 Gas Storage Tanks
3.24 Explosive Gas Mixture

Specifications 3.18.5 and 3.24 provide radioactivity and oxygen limits for the Waste Gas Decay Tanks. The Waste Gas Decay Tanks are used to store fission product gases produced during reactor operation and removed from the RCS by the Makeup and Purification System. The reactor has been defueled, and the RCS depressurized and vented to atmosphere for an extended period of time, such that the there is no hydrogen production mechanism, and there are not appreciable quantities of noble gas present in the RCS and SFP for capture and transfer to the gas decay tanks. Therefore, no Technical Specifications are required for oxygen or radionuclide level monitoring of the gas decay tanks.

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 PCP. The PCP is included for NRC review.

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 REMP. The REMP is included for NRC review.

3.27 Specification 3.27 provides requirements for the Nuclear Service Electrical Building Emergency (NSEB) 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. 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 the building is deenergized. Therefore, there is a significant period of time available to restore the normal or emergency HVAC systems should one of the systems be lost. Operation at power during Cycle 7 was permitted on an interim basis without these systems due to the low heat loads present in the NSEB and the compensatory ventilation measures available. There is no need to have a Specification on these systems since the safety implications of a loss of HVAC in the NSEB are small and require a significant period of time to develop.

3.28 This Specification provides requirements for the TDI Diesel Generator Control Room Essential Ventilation System. The minimum time period available to restore electrical power to the site on a LOOP, 74.4 hours, eliminates the need for emergency diesel generators. Therefore, this Specification is not included in the DTSs.

3.29 This Specification provides requirements for the Meteorological Monitoring Instrumentation. This instrumentation is used for off-site dose calculations and Emergency Preparedness protective action guideline recommendations. The extended decay time of the fuel (207 days as of January 1, 1990) has eliminated the possibility of an off-site dose to a member of the public that would require evacuation. Also, the activity level of gaseous effluent releases from the site has decreased to the point where the default values for offsite dose calculations can be used without impacting releases. The default values are currently used for offsite dose calculations when the Meteorological Monitoring Instrumentation is out of service. An eleven year operating baseline is available for the meteorological data. Therefore, the Meteorological Monitoring Instrumentation is not needed and is not included in the DTSs.

3.30 This Specification provides requirements for the Hydrogen Recombiners. The Hydrogen Recombiners are required to be Operable when the reactor is subcritical by less than 1 percent dk/k. Since the reactor is defueled the Hydrogen Recombiners are not required.

4.0.1 This Specification is included in the DTSs as D4.0.1.

4.0.2 This Specification has been modified for applicability in the Defueled Mode and is included as D4.0.2.

4.0.3 This Specification provides restrictions on entry into an operational Mode. Since the DTSs apply only in the Defueled Mode, this Specification does not apply and is not included.

4.0.4 This Specification provides an extension for certain 18 and 24 month surveillances to allow their performance during the Cycle 8 refueling outage. This Specification is not applicable in the Defueled Mode and is not included in the DTSs.

4.1 This Specification provides testing requirements for instrumentation associated with the Reactor Protection System (RPS), Safety Features System, Process Instrumentation, and Emergency Shutdown Instrumentation. Most of this instrumentation is needed for accident identification and mitigation for an operating nuclear power plant and is not applicable in the Cold Shutdown or Defueled Modes. Several items are applicable in the Defueled Mode as follows:

Table 4.1-1, Items 44a, 44b, 45, and 63 have been dispositioned as follows:

a) Those radiation monitors associated with effluents have been relocated to the ODCM in accordance with the requirements of Generic Letter 89-01.

b) The SFP area radiation monitors are included as DTS D3/4.4.

c) The remaining radiation monitors are used for general area and process monitoring; are not included in the Standard Technical Specifications; are controlled by administrative and technical procedures; and are therefore not included in the DTSs.

Table 4.1-1, Item 44c is not required if the chlorine in the Restricted Area is maintained below 100 pounds. The level of chlorine is procedurally controlled and a Specification is thus not required.

Table 4.1-1, Item 46 has been relocated to the REMP in accordance with Generic Letter 89-01.

Table 4.1-1, Item 47, the 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, the accelerometer would only provide information that is also available from State or Federal Agencies. Therefore, this instrument is not included in the DTSs since adequate time is available to obtain this type of information from other sources.

Table 4.1-1, Item 51, the SFP Level is included as DTS D3/4.1.

Table 4.1-2, Item 5, the requirement to test the refueling system interlocks is included as DTS D3/4.3.

Table 4.1-2, Item 8, see the discussion for Technical Specification 3.13 for the justification for not including this testing requirement.

Table 4.1-2, Item 9, the requirement to test the fire pumps has been relocated to the Fire Protection Plan pursuant to the requirements of Generic Letter 88-12. This relocation was previously docketed in Proposed Amendment No. 180.

Table 4.1-2, Item 11, the requirement to functionally test the SFC system each refueling interval prior to fuel handling is met on a continual basis by the maintenance of the SFP temperature below 140 degrees Fahrenheit. This Specification is intended to test the system prior to the addition of more spent fuel to the SFP. With the constant reduction of the heat load through decay, and the design capability of the SFC system to remove 8.76 million BTU/hour verses the 3.6 million BTU/hour calculated decay heat load on January 3, 1990, the system possesses 240% of the required capacity. By June 7, 1990 the capacity will increase, due to additional spent fuel decay, to 315% of that required. Extraordinary system degradation would have to occur for the system to not meet its intended function. Therefore, no functional test other than verification through normal operation will be performed.

Table 4.1-3, Items 2,4, and 6, the requirements to maintain a minimum boron concentration in the Borated Water Storage Tank, SFP, and Concentrated Boric Acid Tank are not required in the Defueled Mode since borated water is not needed to maintain adequate shutdown margin in the SFP. Therefore, these requirements are not included in the DTSs.

4.2 Specifications 4.2.2.1 through 4.2.2.5 provide inspection, examination, and testing requirements for systems as called out in ASME Section XI for Code Class 1, 2, and 3 components. The inspections identified verify the integrity over time of high temperature, high pressure systems. The systems remaining at Rancho Seco are at relatively low pressure and temperature when compared to their design limits. Therefore, erosion and corrosion expected at high temperature and pressure would not be expected. ASME Section XI, section IWA-2400(c) provides for the deferral of inspections due to extended plant outages. These inspection and examination requirements will not be included in the DTSs.
The testing requirements for pumps and valves are established to verify equipment is capable of meeting their 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 decay of the spent fuel. The pumps and valves required to remain in service will be maintained under the Districts preventive maintenance program which includes vibration monitoring for rotating equipment to identify problems before they lead to failure. This program is similar to that which would be implemented under Section XI. These requirements will not be included in the DTSs.

4.2.3 The monitoring of RCS leakage as required by this Specification is not needed with the reactor defueled and will not be included.

4.3.3 The verification of RCS integrity is not required with the reactor defueled and will not be included in the DTSs.

4.4 Specifications 4.4.1.1 through 4.4.2.6.4 provide testing and inspection requirements for the Reactor Building. With the reactor defueled there are no postulated accidents that would require Reactor Building Integrity. These Specifications will not be included in the DTSs.

4.5 Specifications 4.5.1.1 through 4.5.3.2 provide testing requirements to verify proper operation of emergency core cooling and reactor building spray systems. The reactor core has been defueled and the need for these systems no longer exists. Therefore, these requirements will not be included in DTSs.

4.6 For evaluation see Specification 3.7.

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 and will not be included in the DTSs.

4.8 Specifications 4.8.1 through 4.8.5 provide testing requirements for the Auxiliary Feedwater System. Since the reactor has been defueled there is no need to provide a feedwater supply for the steam generators. Therefore, these Specifications will not be included in the DTSs.

4.9 Specification 4.9 requires verification of reactivity anomaly values during plant operation. Since the core is defueled this Specification is not applicable and will not be included in the DTSs.

4.10 For evaluation see Specification 3.13.
4.11 For evaluation see Specification 3.13.
4.12 For evaluation see Specification 3.13.

4.13 This Specification provides inspection requirements for high energy lines outside the reactor building. Since the high energy systems are not in service in the Defueled Mode, this Specification is not applicable and is not be included in the DTSs.

4.14 For evaluation see Specification 3.12.

4.15 The requirements of this Specification, radioactive material source leakage test, are included as DTS D3/4.7.

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, these requirements will not be included in the DTSs.

4.18 For evaluation see Specification 3.14.

4.19 For evaluation see Specification 3.15.

4.20 For evaluation see Specification 3.16.

4.21.1 For evaluation see Specification 3.17.1.

4.21.2 For evaluation see Specification 3.17.2.

4.21.3 For evaluation see Specification 3.17.3.

4.21.4 For evaluation see Specification 3.17.4.

4.22.1 For evaluation see Specification 3.18.1.

4.22.2 For evaluation see Specification 3.18.2.

4.22.3 For evaluation see Specification 3.18.3.

4.22.4 For evaluation see Specification 3.18.4.

4.22.5 For evaluation see Specification 3.18.5.

4.23 This Specification was previously deleted.

4.24 This Specification was previously deleted.

4.25 For evaluation see Specification 3.21.

4.26 For evaluation see Specification 3.22.

4.27 For evaluation see Specification 3.23.

4.28 For evaluation see Specification 3.24.

4.29 For evaluation see Specification 3.25.

4.30 For evaluation see Specification 3.26.

4.31 For evaluation see Specification 3.27.

4.32 For evaluation see Specification 3.28.

4.33 This Specification was previously deleted.

4.34 For evaluation see Specification 3.29.

4.35 For evaluation see Specification 3.30.

5.1 The Site as described in the Appendix A Technical Specifications has been modified to show the Site Boundary For Liquid Effluents for both 10 CFR 50, Appendix I, and 10 CFR 20 compliance on one drawing for clarity. The Site Figures are included in Defueled Technical Specifications Section D5.0. Non applicable details have been removed from the Figures for clarity.

5.2 This section which describes the Reactor Building has not been included since it is not applicable in the Defueled Mode.

5.3 This section which describes the Reactor has not been included since it is not applicable in the Defueled Mode.

5.4 The descriptions for spent fuel storage in the Spent Fuel Storage Building have been included as DTS D5.2. The remainder of this Specification which addresses new fuel storage is not applicable in the Defueled Mode and is not included.

6.1 The responsibilities called out are redefined for the Defueled Mode and included as Section 6.1 of the DTSs. The titles and responsibilities have been changed to reflect the present descriptions. Directives are in place to delineate responsibility for the current Technical Specification duties.

6.2.1 The organization is now located onsite so no reference to an offsite organization is made. The remainder of the Specification has been changed to reflect new position titles and the Defueled Mode and is included as DTS D6.2.1.

6.2.2 Specification 6.2.2.a is included as DTS D6.2.2. The number of Licensed Operators has been reduced from three in Cold Shutdown to one Licensed Operator on shift and one Senior Licensed Operator on call in the Defueled Mode. This change meets the requirements of 10 CFR 50.54(m)(1).
Specifications 6.2.2.b through 6.2.2.e are related to reactor operation and are not included.
Specification 6.2.2.f defines the Fire Brigade requirements and is removed per Generic Letter 88-12.
Specification 6.2.2.g requires the limiting of overtime for certain personnel. This restriction is generally tied to plant operation since the postulated accidents posing significant risk to plant personnel and health and safety of the public occur during plant operation and not in a Cold Shutdown condition. Therefore, Specification 6.2.2.g is not included.
Specification 6.2.2.h is included as DTS D6.2.2.

6.3.1 This Specification is included as DTS D6.3, except the requirements for a Shift Technical Advisor (STA) have been deleted since an STA is not required in the Defueled Mode.

6.4.1 This Specification has been modified to reference the NRC approved training program and is included as DTS D6.4.

6.4.2 This Specification for Fire Brigade training was removed per Generic Letter 88-12.

6.5.1 The requirements for a Plant Review Committee have been modified to reflect the activities of a defueled plant and have been included as DTS D6.5.1.

6.5.2 The requirements for a Management Safety Review Committee have been modified to reflect the activities of a defueled plant and have been included as DTS

D6.5.2.

6.5.3 The requirements of this Specification are included in DTS D6.5.3.

6.5.4 The requirements of this Specification have been included in DTS D6.5.4, with the following changes in Audit frequencies:

The Facility Emergency Plan audit frequency is changed to yearly from two years to reflect the audit performed in accordance with 10 CFR 50.54(t).
The Facility Security Plan audit frequency is changed to yearly from two years to reflect the audit performed in accordance with 10 CFR 73.55(g)(4).
The audit of the REMP is changed to 24 months from 12 months to be consistent with the audit frequencies for the ODCM and PCP. The reduction in audit frequency for the REMP is consistent with the reduction in actual offsite releases associated with a defueled plant. This reduced audit frequency does not alter the requirement to maintain the control and monitoring of offsite releases.

6.6.1 This Specification, Licensee Event Reports, is included as DTS D6.6.

6.7 This Specification provides actions should a Safety Limit be violated. Since there are no Safety Limits in the DTSs, this Specification is not included.

6.8 This Specification has been modified to reflect those procedures required for a defueled plant and is included as DTS D6.8. In addition, requirements for the radioactive effluents controls program and the REMP are also included in accordance with Generic Letter 89-01.

6.9.1 The Specification provides requirements for submittal of the Startup Report. This is not applicable for a defueled plant and is not included.

6.9.2.1.1 This Specification, Annual Occupational Radiation Exposure Report, is included as DTS D6.9.1.1.

6.9.2.1.2 This Specification, Annual Exposure Report, is included as DTS D6.9.1.2.

6.9.2.2 This Specification, Annual Radiological Environmental Operating Report, was modified to meet the requirements of Generic Letter 89-01 and included as DTS D6.9.1.3.

6.9.2.3 This Specification, Semiannual Radioactive Effluent Release Report, was modified to meet the requirements of Generic Letter 89-01 and included as DTS D6.9.2.

6.9.3 This Specification provides the requirements for the contents of the Monthly Report. This report deals primarily with the operating statistics for the plant. This has been changed to an Annual Report, which is consistent with the requirements of 10 CFR 50.59(b)(2), and is reflective of the significant reduction in plant activities anticipated for a defueled reactor.

6.9. This Specification provides requirements on reportability pursuant to 10 CFR 50.73. The detailed requirements have been removed and replaced with a reference to meet 10 CFR 50.73 to avoid a conflict with potential future changes to the Code. This is included as DTS D6.9.4.

6.9.5 The Special Reports required by this Specification were not included for the following reasons:

6.9.5.A was a one time only report for 1977.
6.9.5.B,C,D,F, and P are associated with an operating plant only.
6.9.5.E was removed per Generic Letter 88-12.
6.9.5.G,H,I,J,K,M,N, and O have been removed per Generic Letter 89-01.
6.9.5.L was previously deleted.

6.9.6 This Specification, Environmental Reports, has been included as DTS D6.9.5.

6.10 The requirements for record retention have been included as DTS D6.10. The following changes were made:

The following deletions were made:

6.10.1.e relates to reactor tests and experiments and therefore is not applicable.
6.10.2.e relates to in-service inspections and, since the plant is not in service, this is not applicable.
6.10.2.1 relates to environmental qualification and is not applicable since the accidents that result in a harsh environment are not credible in the Defueled Mode.

In addition, the following requirements were added:

Retention of records of reviews performed for changes made to the ODCM, REMP MANUAL, and the PCP was added in

accordance with Generic Letter 89-01.
Retention of records related to decommissioning activities.

6.11 This Specification is included as DTS D6.11.

6.12 Previously deleted.

6.13 The requirements for control of High Radiation Areas is included as DTS D6.12.

6.14 This Specification, Environmental Qualification, is not applicable in the Defueled Mode since the accidents that result in a harsh environment are not credible.

6.15 This Specification, PCP, was revised to conform to the guidelines of Generic Letter 89-01 and included as DTS D6.13.

6.16 This Specification, ODCM and REMP, was revised to conform to the guidelines of Generic Letter 89-01 and included as DTS D6.14.

6.17 This Specification, Major Changes to Radioactive Waste Treatment Systems, was removed in accordance with guidelines of Generic Letter 89-01. Changes to the radioactive waste treatment systems are controlled under the PCP, and are reviewed by the Plant Review Committee.

6.18 This Specification, Postaccident Sampling, is not applicable in the Defueled Mode.

The requirements of Appendix B to Operating License NO. DPR-54 have been previously justified for deletion from Technical Specifications in Proposed Amendment No. 102. Therefore, they are not included in the DTSs.

TABLE 1

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

SUMMARY

The Rancho Seco DTSs provide the controls necessary for protection of health and safety of the public in the defueled condition. These DTSs 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 duration of the current extended outage with the reactor defueled.

The DTSs were developed through an evaluation of credible accidents, and equipment needed to mitigate these accidents, in the Defueled Mode. Two accidents are considered credible in the Defueled Mode; 1) a fuel handling accident, and 2) a LOOP. Based on maintaining the radioactivity release levels to as low as reasonably achievable, the Rancho Seco Technical Specifications were reviewed and the appropriate requirements incorporated in the DTSs. A detailed description of this evaluation is provided in this safety analysis.

The DTSs as presented in Proposed Amendment 182 provide technical and administrative controls sufficient to assure protection of health and safety of the public. They do not result in a significant decrease in the margin of safety provided by the Appendix A and B Technical Specifications.

NO SIGNIFICANT HAZARDS CONSIDERATION

The District has reviewed the proposed DTSs against each of the criterion of 10 CFR 50.92 and concluded that the proposed changes as described 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 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 not changed since the fuel handling accident scenario remains unchanged. The extended time period available to restore offsite power (9.18 days minimum as of January 3, 1990) provides ample time to take corrective action to ensure fuel damage does not occur on a loss off 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 of the DTSs applies only when the reactor is defueled. Therefore, only those activities and potential accidents associated with the SFP need be considered. Since the physical characteristics of the plant associated with the SFP are not being changed, and the systems required to safely store the spent fuel will 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 for the two credible accidents, the margin of safety for the fuel handling accident is unchanged, while the margin of safety for the LOOP is not significantly reduced given the 9.18 day minimum time period available to restore offsite power as of January 3, 1990, 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 changes do not constitute a significant hazard to the public and do not endanger the public's health and safety.

REFERENCES

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

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

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

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

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

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

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

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

k. NRC to SMUD letter dated September 13, 1982, TECHNICAL SPECIFICATIONS (TSs) FOR EMERGENCY SAFETY FEATURE (ESF) AIR FILTERS

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

L. Regulatory Guide 1.155, STATION BLACKOUT, June 1988

M. Rancho Seco Nuclear Generating Station, Unit No. 1, Updated Safety Analysis Report, Amendment 6

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

ATTACHMENTS

1. OFFSITE DOSE CALCULATION MANUAL
2. RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM MANUAL
3. PROCESS CONTROL PROGRAM