October 16, 1974

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

PACIFIC GAS AND ELECTRIC COMPANY

DIABLO CANYON NUCLEAR POWER STATION, UNITS 1 AND 2

SAN LUIS OBISPO COUNTY, CALIFORNIA

DOCKET NOS. 50-275 AND -323

3/87
Reg

TABLE OF CONTENTS

BLANK PAGE

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

APPENDICES

TABLES

FIGURES

ABBREVIATIONS

a-c	alternating current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	United States Atomic Energy Commission
AISC	American Institute of Steel Construction
ALAP	as low as practicable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transients without scram
BEA	Bureau of Economic Analysis
BIT	boron injection tank
BOL	beginning of life
BRS	boron recycle system
BTU	British Thermal Units
BTU/hr	British Thermal Units per hour
BTU/hr-ft^2	British Thermal Units per hour per square foot
BWR	boiling water reactor
CA	wake factor

cal/gm	calories per gram
CAM	continuous air monitor
cc	cubic centimeter
CEA	control element assembly
cfm	cubic feet per minute
cfs	cubic feet per second
CFR	Code of Federal Regulations
Ci	Curies
Ci/yr	Curies per year
cm/sec	centimeters per second
CO_2	carbon dioxide
CP	construction permit
CVCS	chemical and volume control system
DBA	design basis accident
d-c	direct current
DF	decontamination factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOT	Department of Transportation
$\Delta k, \Delta\rho$	reactivity change
ΔT	temperature change or difference
ECCS	emergency core cooling system
EOL	end of life
ESF	engineered safety features

ESFAS	engineered safety feature actuation systems
°F	degrees Farenheit
FLECHT	full length emergency cooling heat transfer
F_Q	peaking factor
FSAR	Final Safety Analysis Report
ft	feet
FES	Final Environmental Statement
ft^2	square feet
ft^3	cubic feet
ft/sec	feet per second
g	gravitational acceleration, 32.2 feet per second per second
GDC	AEC General Design Criteria for Nuclear Power Plants
G-M	Geiger-Mueller
GONPR & AC	General Office Nuclear Plant Review and Audit Committee
gpd	gallons per day
gpm	gallons per minute
GWPS	gaseous waste processing system
HEPA	high efficiency particulate air
hrs	hours
IEEE	Institute of Electrical and Electronics Engineers
in	inch
km	kilometer
kV	kilovolt
kW	kilowatt
kWh	kilowatt-hours
kW/ft	kilowatts per foot
lb	pound

lb/hr-ft²	pounds per hour-square foot
lb/ft³	pounds per cubic foot
lb/hr	pounds per hour
LOCA	loss-of-coolant accident
LPD	linear power density
LPZ	low population zone
LWTS	liquid waste treatment system
m	meter
m²	square meters
mph	miles per hour
m/sec	meters per second
MLLW	mean lower low water
MSL	mean sea level
MW	megawatts
MWD/MT	megawatt-days per metric ton
MWe	megawatts electrical
MWt	megawatts thermal
mrad	one thousandth of a rad
mrem	one thousandth of a rad equivalent man
mrem/yr	one thousandth of a rad equivalent man per year
NaOH	sodium hydroxide
NDTT	nil ductility transition temperature
NPSH	net positive suction head
NSSS	nuclear steam supply system
nvt	neutron fluence, neutrons per square centimeter

OBE	operating basis earthquake
PG&E	Pacific Gas and Electric Company
pH	expression of acidity or alkalinity on a scale of 0-14
PMF	probable maximum flood
PMP	probable maximum precipitation
PNAC	President's Nuclear Advisory Committee
ppm	parts per million
PSAR	Preliminary Safety Analysis Report
psf	pounds per square foot
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PSRC	Plant Staff Review Committee
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
RCS	reactor coolant system
RHR	residual heat removal
RHRS	residual heat removal system
RCPB	reactor coolant pressure boundary
RPS	reactor protection system
rem	rad equivalent man
R & D	research and development
rpm	revolutions per minute

RWST	refueling water storage tank
scfm	standard cubic feet per minute
sec/m^3	seconds per cubic meter
SER	Safety Evaluation Report
SGBTS	steam generator blowdown treatment system
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection system
SSE	safe shutdown earthquake
std	standard
TBS	turbine bypass system
TDC	thermal diffusion coefficient
TMD	Transient Mass Distribution
U-235	uranium 235
U-238	uranium 238
UO_2	uranium dioxide
USAS	United States of America Standard
USGS	United States Geological Survey
v/o	volume percent
w/o	weight percent
wg	water gauge
X/Q	relative concentration
yr	year
10 CFR	AEC, Title 10, Code of Federal Regulations
Part 1	Statement of Organization and General Information
Part 2	AEC Rules of Practice

Part 20	AEC Standards for Protection Against Radiation
Part 50	AEC Licensing of Production and Utilization Facilities
Part 55	Operators' Licenses
Part 71	Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions
Part 73	Physical Protection of Special Nuclear Material
Part 100	AEC Reactor Siting Criteria
Part 140	Financial Protection Requirements and Indemnity Agreements
Part 170	Fees for Facilities and Materials Licenses Under the Atomic Energy Act of 1954, as Amended

1.0 INTRODUCTION

1.1 General Background

The Pacific Gas and Electric Company (PG&E, and hereinafter referred to as the applicant) filed with the Atomic Energy Commission (Commission) applications dated January 16, 1967 and June 28, 1968, and as subsequently amended, for licenses to construct and operate pressurized water reactors, identified as Units 1 and 2, respectively, of the Diablo Canyon Nuclear Power Station. These Units are being constructed at the Diablo Canyon site which is located on the central California coast in San Luis Obispo County, approximately twelve (12) miles west-southwest of San Luis Obispo, the County Seat. Unit 1 is being constructed under AEC Construction Permit CPPR-39 issued on April 23, 1968, and Unit 2 under CPPR-69 issued on December 9, 1970. On July 10, 1973, the applicant filed an application for operating licenses to operate Units 1 and 2 of the Diablo Canyon Plant. Included as part of this application was a Final Safety Analysis Report (FSAR) as required by 10 CFR 50.34(b). The application for operating licenses was docketed on October 2, 1973.

The application requests operating licenses of 3338 and 3411 thermal megawatts (MWt) for Units 1 and 2, respectively; these levels are equivalent to net electrical outputs of 1084 and 1106 electrical megawatts (MWe) for Units 1 and 2, respectively. The slight difference in output for the two units is due to the upgraded turbine generator design for Unit 2. The Unit 1 thermal power level (3338 MWt) is slightly higher

than the 3250 value given in the Preliminary Safety Analysis Report (PSAR) which was tendered in 1967. The applicant states that the expected ultimate thermal outputs for Units 1 and 2 are 3488 and 3568 MWt, respectively; the corresponding net electrical outputs for these values are 1131 and 1156 MWe.

The radiological safety review with respect to a decision concerning issuance of operating licenses for Diablo Canyon Units 1 and 2 has been based on the applicant's Final Safety Analysis Report and subsequent Amendments 1 through 17, all of which are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, N.W., Washington, D. C., and at the San Luis Obispo County Library, 888 Morrow Street, San Luis Obispo, California 93406.

During the course of the safety review of the material submitted, we held a number of meetings with representatives of the applicant, his consultants, and Westinghouse Electric Corporation to discuss the plant design, construction, proposed operation and performance under postulated accident conditions. During our review, we requested the applicant to provide additional information that we needed for our evaluation. This additional information was provided in Amendments to the application. As a result of our review, a number of changes were made in the facility design and proposed operating practices; these changes are described in the applicant's Amendments to the FSAR and are discussed in appropriate sections of this report. Section 1.6 provides a listing of the principal design changes which were made. A chronology of the principal actions

relating to the processing of the application is attached as Appendix A to this Safety Evaluation Report (SER).

This Safety Evaluation Report summarizes the results of the radiological safety review of Diablo Canyon Units 1 and 2 that was performed by the Commission's Regulatory staff. The review and evaluation of the facilities for operating licenses is only one stage in the continuing review by the staff of the design, construction, and operating features of Units 1 and 2. The proposed design of these facilities was reviewed before construction permits were issued. Construction of these facilities has been monitored in accordance with the inspection program of the Commission's Regulatory staff. At this, the operating license application phase, we have reviewed the final design to determine that all of the Commission's safety requirements have been met. If operating licenses are granted, the facilities will be operated only in accordance with the terms of the operating licenses and the Commission's regulations, and will be subject to the continuing inspection program of the Regulatory staff.

In addition to our review, the Advisory Committee on Reactor Safeguards (ACRS) is reviewing the application and has met and will meet with both the applicant and the Regulatory staff to discuss the facilities. The ACRS report to the Commission on the Unit 1 and 2 facilities will be provided in a supplement to this Safety Evaluation Report.

The conclusions reached as a result of our evaluation of PG&E's application to operate Diablo Canyon Units 1 and 2 are presented in Section 22 of this Safety Evaluation Report.

1.2 General Plant Description

Units 1 and 2 of the Diablo Canyon Nuclear Power Station are located on a 750 acre site in San Luis Obispo County, California. The site is adjacent to the Pacific Ocean and is roughly equidistant from San Francisco and Los Angeles.

Units 1 and 2 are substantially identical pressurized water nuclear power units, each consisting of a Nuclear Steam Supply System (NSSS) in a 4-loop reactor coolant system, turbine generator, auxiliary equipment, and controls and instrumentation. For each unit, the principal structures include the containment, the turbine building, and the auxiliary building (which includes the control room, the fuel handling areas and the ventilation areas). The ultimate heat sink for rejection of heat from the Diablo Canyon Units is the Pacific Ocean.

The NSSS for each unit consists of a pressurized water reactor, reactor coolant system, and associated auxiliary fluid systems. The reactor coolant system consists of four parallel reactor coolant loops, each containing a steam generator and a reactor coolant pump. A pressurizer is connected to the hot leg of one reactor coolant loop. The basic fuel design is a 17x17 matrix of fuel rods in each fuel assembly.

The reactor core is composed of an array of 193 fuel assemblies, each containing 264 fuel rods. These rods are composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. All fuel rods are pressurized with helium during fabrication to reduce stress and increase fatigue life. The reactor core will initially contain 3 regions of slightly different enrichments of U-235.

The reactor will be controlled by control element movement and regulation of the boron concentration in the reactor coolant. The full length rod cluster control assemblies are stainless steel tubes containing a silver-indium-cadmium absorber, and are positioned by drive mechanisms of the magnetic latch type. Part length control rods are also provided for use in controlling axial power distribution. A soluble poison (boron) is introduced into the reactor coolant to compensate for long term reactivity changes.

The reactor vessel and reactor internals contain and support the fuel and rod cluster control assemblies. The vessel is cylindrical with hemispherical heads and is clad with stainless steel.

The pressurizer is a vertical cylindrical pressure vessel with hemispherical heads and is equipped with electrical heaters and spray nozzles for system pressure control. The steam generators are vertical U-tube type heat exchangers with Inconel tubes. Reactor coolant flows inside the tubes; steam is generated in the shell and flows through the main steam lines to the turbine. Integral moisture separating equipment reduces moisture content of the steam at the turbine throttle to 0.25 percent or less. The reactor coolant pumps are vertical, single-stage, centrifugal units equipped with controlled leakage shaft seals.

Auxiliary systems are provided to charge the reactor coolant system and add makeup water, to purify reactor coolant water, to provide chemicals for corrosion inhibition and reactor control, to cool system components, to remove residual heat when the reactor is shut down, to

cool the spent fuel storage pool, to sample reactor coolant water, to provide for emergency safety injection, and to vent and drain the reactor coolant system.

The engineered safety features provided for the Diablo Canyon Units have sufficient capacity and redundancy to protect the health and safety of the public by keeping exposures below the limits set forth in 10 CFR Part 100 for any postulated malfunction or accident, including the most severe loss of coolant accident.

An instrumentation and control system provides automatic protection against unsafe and improper reactor operation during both steady state and transient conditions. The entire operation of the plant is monitored and controlled by operators in the control room, which is located in the auxiliary building.

The electrical systems generate and transmit power to the applicant's high voltage system, distribute power to the auxiliary loads, and provide control, protection, instrumentation and annunciator power supplies for the units. Offsite a-c power is available from two 230 kV transmission lines and three 500 kV transmission lines. The 230 kV line serves the standby/startup transformers for both Units 1 and 2. Onsite a-c emergency power is supplied by redundant and independent diesel generators. Two diesel generators are dedicated to each unit and a fifth generator can serve either unit. Onsite d-c power for each unit is available from three 125 v batteries.

The steam and power conversion system is designed to receive the heat absorbed by the reactor coolant system during normal power operation, as well as following an emergency shutdown of the turbine generator from full load. Heat rejection under the latter condition is accomplished by steam bypass to the condenser and pressure relief to the atmosphere.

Auxiliary systems are supporting systems included in the facility, some of which are required to perform certain functions during emergency or accident conditions. Included are the cooling water systems, the heating and ventilating systems, the fire protection system, the process auxiliaries, the compressed air system, the diesel generator fuel oil system, the communication systems, and the lighting systems.

Certain facilities and equipment are shared between Units 1 and 2. In terms of structures, the two units share a common auxiliary building where the major portion of the radioactive waste treatment equipment is shared by the two units. Also, the plant is provided with a central control room located in the auxiliary building. Physical separation of control panels eliminates interaction of Unit 1 and 2 control systems. The turbine building for Unit 2 is an extension of the Unit 1 building. The two units also share a common raw water storage reservoir, fire pumps, fire water storage tank, diesel fuel oil storage tanks and transfer pumps, auxiliary boiler, makeup water system, plant air system and lubricating oil storage system. Details of the sharing of these systems are discussed in appropriate sections of this report.

1.3 Comparison with Similar Facility Designs

Many features of the design of the Diablo Canyon Units 1 and 2 are similar to those that we have evaluated and approved previously for other nuclear power plants now under construction or in operation, particularly the Zion Units 1 and 2 (Docket Nos. 50-295 and 50-304). The principal difference between the Diablo Canyon and Zion Units is the 17x17 fuel assembly design which is planned for Diablo Canyon. In addition to Zion, the Diablo Canyon Units are quite similar to the Trojan Nuclear Plant (Docket No. 50-344), whose operating license application is currently under review by the staff. To the extent that is feasible and appropriate, we have made use of these previous evaluations in conducting our review of the Diablo Canyon Units. Our Safety Evaluation Reports for these other facilities that have been approved have also been published and are available for public inspection at the Atomic Energy Commission's Public Document Room in Washington, D. C.

1.4 Identification of Agents and Contractors

Pacific Gas and Electric Company (PG&E) is the sole applicant for operating licenses for Units 1 and 2 of the Diablo Canyon Nuclear Power Station. PG&E is the architect-engineer, constructor, operator, and owner of the Diablo Canyon Nuclear Power Station, and as such assumes full responsibility and authority for the design, construction, startup, and operation of Units 1 and 2. The applicant has engaged the services of Westinghouse Electric Corporation (hereinafter referred to as Westinghouse) to design and manufacture the NSSS for both Units.

Westinghouse is also responsible for fabrication of the nuclear fuel for the initial cores for Units 1 and 2. In addition to Westinghouse as the contractor for the NSSS, the applicant has retained numerous consultants who performed investigations and submitted reports to PG&E on a wide range of subjects. A list of these consultants and their subject areas is provided in Table 1.4-1 of the FSAR.

1.5 Summary of Principal Review Matters

Our evaluation included a technical review of the information submitted by the applicant, particularly with regard to the following principal matters:

(1) We evaluated the population density and land use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to establish that these characteristics have been determined adequately and have been given appropriate consideration in the final design of the plant, and that the site characteristics are in accordance with the Commission's siting criteria (10 CFR Part 100), taking into consideration the design of the facilities including the engineered safety features provided.

(2) We evaluated the design, fabrication, construction, and testing and performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accordance with the Commission's General Design Criteria (GDC), Quality Assurance Criteria, Regulatory Guides and other appropriate rules, codes

and standards, and that any departures from these criteria, codes and standards have been identified and justified.

(3) We evaluated the expected response of the facilities to various anticipated operating transients and to a broad spectrum of accidents, and determined that the potential consequences of a few highly unlikely postulated accidents (design basis accidents) would exceed those of all other accidents considered. We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite radiation doses that might result in the very unlikely event of their occurrence would not exceed the Commission's guidelines for site acceptability given in 10 CFR Part 100.

(4) We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operations, including the proposed organization, staffing and training program, the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public, to determine that the applicant is technically qualified to safely operate the units.

(5) We evaluated the design of the systems provided for control of the radiological effluents from the plant to determine that these systems are capable of controlling the release of radioactive wastes from the facilities within the limits of the Commission's regulations, and that the equipment provided is capable of being operated by the applicant in such a manner as to reduce radioactive releases to levels that are as low as practicable.

(6) We evaluated the financial position of the applicant to determine that the applicant is financially qualified to operate Diablo Canyon Units 1 and 2.

1.6 Facility Modifications Required as a Consequence of Regulatory Staff Review

As a consequence of the staff review, a number of design changes were or will be made to Units 1 and 2. These modifications are discussed in greater detail within the body of this Safety Evaluation Report. The principal changes are as follows:

(1) Upgrading of the onsite meteorological program with regard to control room monitoring of certain parameters (see Section 2.3.3).

(2) Upgrading of the post-tornado availability of certain Category I systems and structures (see Section 3.5).

(3) Design measures taken for protection against the dynamic effects associated with pipe ruptures outside containment (see Section 3.6).

(4) Augmentation of the seismic instrumentation program (see Section 3.7).

(5) Installation of a loose parts monitoring system (see Section 5.4).

(6) Installation of appropriate chlorine protection devices in the control room (see Section 6.4).

(7) Improvement of the response time testing program for components of the reactor protection system (see Section 7.2.4).

(8) Provision for automatic opening of accumulator isolation valves when reactor coolant pressure exceeds a preselected value (see Section 7.3.3).

(9) Provision for automatic tripping of the residual heat removal pumps when the water in the refueling water storage tank reaches a specified low level (see Section 7.3.4).

(10) An increase in the diversity of interlocks for the residual heat removal system to prevent possible over-pressurization of this system (see Section 7.6).

(11) Tentative commitment that spent fuel will not be stored in the spent fuel pool in locations where it could be struck by a dropped cask (see Section 9.2.3).

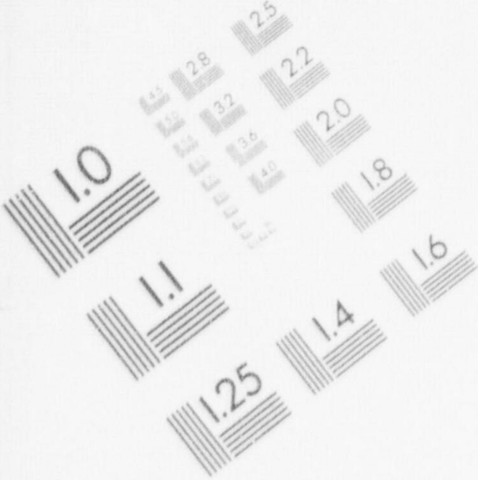
(12) Installation of expansion joint sleeves around each expansion joint in the circulating water system to preclude flooding of safety related equipment in the turbine building (see Section 10.4).

(13) Commitment to provide a flood wall in the turbine building to prevent flooding in the event of a water box failure in the circulating water system (see Section 10.4).

(14) Revision of the operator requalification program (see Section 13.2).

(15) Requirements for arming of the plant security guards (see Section 13.6).

(16) Modification of the QA program for operations to provide for an independent qualified inspection staff reporting to the QA Director (see Section 17.3).



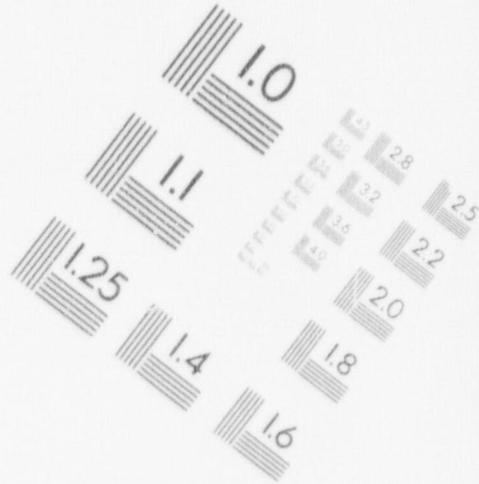
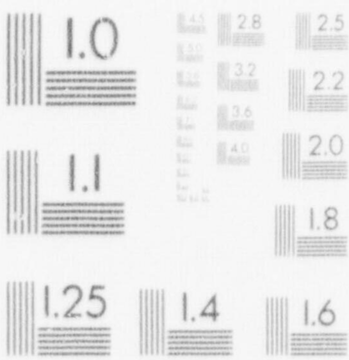
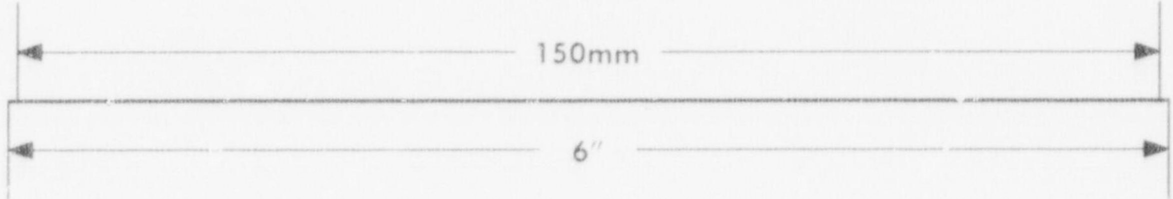
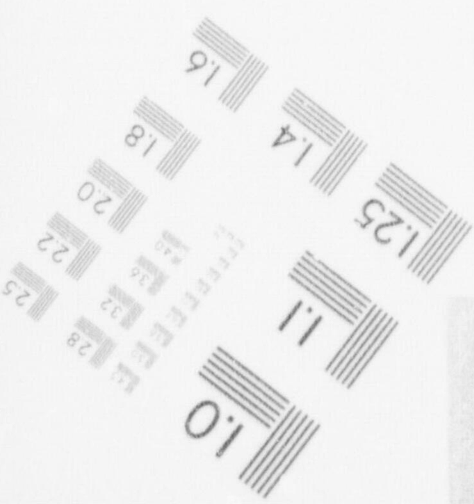


IMAGE EVALUATION TEST TARGET (MT-3)







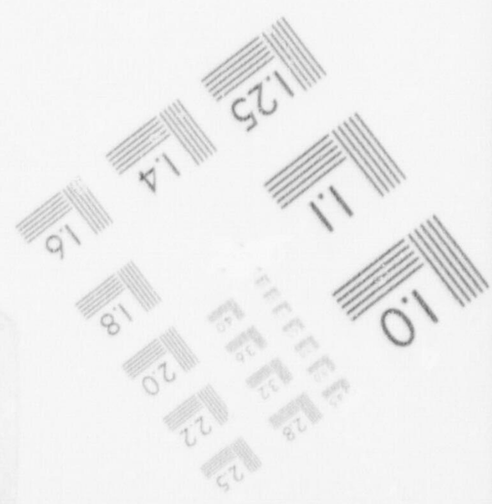


IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location

The Diablo Canyon site is adjacent to the Pacific Ocean in San Luis Obispo County, California, and is approximately 12 miles west-southwest of the city of San Luis Obispo, the County Seat. The site is roughly equidistant from San Francisco and Los Angeles. Figure 2.1 locates the site on a map of west-central California.

2.1.2 Site Description

The site consists of approximately 750 acres near the mouth of Diablo Creek. The parcel immediately south of the creek consists of 585 acres and has been leased to the applicant for a term of 99 years with an option to renew for an additional 99 years. The 165 acre parcel on the north side of the creek is owned by the applicant. The site boundary and the location of principal structures are shown in Figure 2.2. The locations of the gaseous and liquid effluent release points are also given. As shown in Figure 2.2, a portion of the site is bounded by the Pacific Ocean. The distance from either reactor to the nearest site boundary on land is one-half mile (approximately 800 meters) which is the exclusion distance. The minimum distance from either reactor to the ocean (mean high water) is 600 feet (approximately 200 meters).

On land there are no activities unrelated to plant operation within the exclusion area. The exclusion area is not traversed by public highway or railroad. The applicant has stated in the FSAR that within the 585 acre parcel leased to the Company, it has the right to use

excavated materials in the construction of the plant. All other mineral rights are retained by the lessor, although the applicant stated that these retained rights cannot be exercised in a manner inconsistent with the applicant's use of the land.

The offshore area (below the mean high water line) within the exclusion distance is not under the applicant's routine control, and is at times entered by commercial or sports fishing boats. The shoreline of the site is rough and precipitous and the land area below the mean high water line could be occupied only with great difficulty. Portions of the exclusion area which extend offshore will be controlled by the U.S. Coast Guard in the event of an emergency.

2.1.3 Population and Population Distribution

Population data taken from the applicant's FSAR have been compared with the 1970 Census and with projections prepared by the Bureau of Economic Analysis (BEA), U.S. Department of Commerce. The nearest residence is 1-3/4 miles north-northwest of the site and is occupied by two persons. The applicant has selected a distance of 6 miles to be the low population zone as defined in 10 CFR Part 100. The 1970 population within this distance is estimated to be 18. We have investigated the possibility of greatly increased residential growth within the low population zone. There is a possibility that a large condominium development, including a resort hotel with recreational facilities, will be located within the low population zone at a distance of one to six miles south of the plant. The maximum number of residents anticipated for this development

would be 2760 persons. Plans for this development, still tentative, call for completion by 1984 and require approval both by the San Luis Obispo County Planning Commission and the California Coastal Environment Commission. The population center (as defined in 10 CFR Part 100) has been selected to be the city of San Luis Obispo, a distance of 12 miles away from the plant, with a 1970 population of 28,036. The distance to the nearest boundary of the population center is more than one and one-third times the low population zone radius, in compliance with 10 CFR Part 100 guidelines. Figure 2.3 shows the 1970 cumulative population surrounding the plant as a function of distance from the plant out to 50 miles. The cumulative population corresponding to a moderately populated area of 400 people per square mile is also shown. Comparison of the curves in the figure shows that the site area is not heavily populated.

U.S. Census data indicate that the city of San Luis Obispo showed an increase of 37% in population from 1960 to 1970, while the county of San Luis Obispo showed a gain of 30% over the same period. For the area within a 50 mile radius of the plant, the applicant projects a population growth of 65% in the 20 year period from 1970 to 1990 and a 148% increase in the 40 yr. period from 1970 to 2010. We have compared the applicant's projections with those prepared by the BEA, U.S. Department of Commerce, for Water Resources SubArea No. 1807. Figure 2.4 shows a circle of 50 mile radius around the Diablo Canyon site in relation to Water Resources SubArea No. 1807. The BEA projects a population growth of 40% for this

area over the 20 year period from 1970 to 1990, and an 83% growth over the 40 year period from 1970 to 2010.

In addition to the resident population, there is a seasonal influx of transient vacation and weekend visitors, primarily to the beach areas, and especially during the summer months. Within the low population zone the maximum number of persons at a single time is estimated to be 5000. This corresponds to the maximum daytime use of the Montana de Oro State Park, whose area of principal use is along the beach, between 4 and 5 miles northwest of the site. Overnight use is considerably less, an estimated maximum of 400 persons. We concur with the applicant that evacuation of these numbers of persons from the park could be accomplished expeditiously and without injury in the event of an emergency.

On the basis of the Part 100 definitions of the population center distance (12 miles), the exclusion radius (one-half mile), and LPZ distance (6 miles), our analysis of the onsite meteorological data from which dilution factors were calculated (Section 2.3 of this report), and the calculated potential radiological dose consequences of design basis accidents (Section 15.0 of this report), we have concluded that the exclusion area radius and the LPZ distance are acceptable. The site meets the requirements of 10 CFR Part 20 with respect to the restricted area.

2.1.4 Uses of Adjacent Lands and Waters

The San Luis Range, which attains a height of approximately 1800 feet, dominates the region between the site and U.S. Route 101. This upland country is used to a limited extent for grazing beef cattle and, to a very minor extent, dairy cattle.

San Luis Obispo County has relatively little level land, except for a few small coastal valleys where farming is the predominant activity. Principal crops include vegetables, poultry, and grain. The county's leading agricultural product is livestock, which constituted over 40 percent of the gross value of farm products sold in 1970.

The nearest dairying activity is between 7 and 8 miles northeast of the site. There are two small operations yielding a total of approximately 1500 gallons per day. One additional small dairy located about 10 miles east of the site produces about 800 gallons per day. The largest dairy within a 15 mile radius is some 12 miles north of the site, and produces 2200 gallons per day.

The ocean area adjacent to the plant is a small part of the larger coastal fishing grounds extending from slightly north of Point Buchon to Point San Luis. In 1966 the total annual combined sport and commercial catch from this fishery was estimated to be 621,000 pounds of abalone, 81,000 pounds of rockfish, and 21,000 pounds of other species. Diablo Cove, where the circulating water discharge structure is located, is estimated to have contributed, in its undisturbed state, about one percent of the fish catch mentioned above.

There are no public water supply ground water basins within 10 miles of the site. Property owners north and south of the site capture surface water from small intermittent streams and springs for private domestic use.

2.2 Nearby Industrial, Transportation and Military Facilities

On the basis of information supplied by the applicant, there are no airports, military or industrial facilities, gas pipelines, major highways or railroads within 5 miles of the Diablo Canyon site.

Significant transportation routes in the vicinity include U.S. Highway 101 which passes about 10 miles east of the site, State Route 1 which passes about 10 miles north of the site, and the Southern Pacific Railroad which roughly parallels U.S. Highway 101 and passes about 10 miles east of the site. The nearest operational airport is the San Luis Obispo County airport located 12 miles east of the site. Aside from fishing vessels, the nearest marine traffic is that associated with local coastal tankers approaching no closer than 5 to 10 miles of the plant to load and unload petroleum products. These products are stored at Avila Beach and Morro Bay, which are located at distances of 8 and 10 miles from the site, respectively.

The largest industrial complex near the site is Vandenberg Air Force Base, located about 35 miles south of the site. Missiles fired to the Western Pacific Missile Range from this base are not directed toward Diablo Canyon.

Because of the absence of industrial, transportation or military facilities in the area of the site, we conclude that safe operation of the plant at the Diablo Canyon site will not be adversely affected.

2.3 Meteorology

2.3.1 Regional Climatology

The climate of the Diablo Canyon site, located along the central coast of California about 200 miles west-northwest of Los Angeles, is maritime, resulting in modified temperature extremes and high humidities. The central coast of California is along the eastern portion of the Pacific high pressure system, which generally dominates the circulation patterns of the region and keeps the principal tracks of low pressure systems north of the area. The Pacific high pressure system weakens and moves southward in the winter, allowing Pacific cold fronts and storm centers to move inland. This results in the well-defined rainy season from November to March.

2.3.2 Local Meteorology

Meteorological data from several local areas have been examined. These areas are the plant site, the city of San Luis Obispo (12 miles east of the site), Oceana (16 miles southeast of the site), and Santa Maria (29 miles southeast of the site). Mean monthly temperatures in this locale may be expected to range from about 50°F in January to about 62°F in July, August, and September. Annual average precipitation is about 16 inches, with about 85% of this occurring from November to March. Wind data from the 25-foot level on the "E" tower at the Diablo Canyon site for the period July 1967 to October 1969 indicate a predominance of winds from the northwest (25%) and from the west-northwest (18%). The prevailing wind direction at Santa Maria for a 21 year

period of record is west-northwest. The mean wind speed at the plant site is about 11 miles per hour (mph), and at Santa Maria this value is 7 mph.

Severe weather is not common in the area. Thunderstorms can be expected on only about 3 days per year. One of the most severe tropical storms on record along the Southern California coast occurred on September 24-25, 1939. This storm moved north along the coast and curved inland near Los Angeles. The most severe effects were observed in the Los Angeles area and south. The storm had little effect on the site. The number of days having high air pollution potential averages about 9 per year at the site.

2.3.3 Onsite Meteorological Measurements Program

Topography at the site is extremely complex, and it is difficult to document representative atmospheric dispersion characteristics. Six meteorology towers have been used by the applicant in order to represent site meteorology. The 250-foot "E" tower will remain as the permanent facility. This tower is located on a relatively flat plain between the major plant structures and the coastal bluff. Instrumentation on this tower consists of the following: (1) temperature sensors at the 25-, 150-, and 250-foot levels; (2) measurement of wind speed and direction at the 25- and 250-foot levels; (3) bivanes measuring horizontal and vertical wind fluctuations at the 25- and 250-foot levels; and (4) measurement of dewpoint temperature at the 25-foot level. The primary data recording system utilizes magnetic tape, with strip charts as the secondary system.

The applicant has presented in Section 2.3.3 of the FSAR a proposed program for control room monitoring of meteorological parameters. This program is currently being reviewed, but the applicant must provide more detailed information before the evaluation can be completed. Resolution of this item will be reported on in a supplement to this Safety Evaluation Report.

The applicant has submitted joint frequency distributions of wind speed and direction by atmospheric stability (as defined by vertical temperature gradient between the 25- and 250-foot levels) for the 25- and 250-foot levels. Data for both of these levels have been supplied for the period July 1967 to October 1969. Additional joint frequency distributions of wind speed and direction by atmospheric stability (as defined by vertical angle fluctuations) for the 25-foot level for the periods October 1969 to March 1971 and April 1972 to September 1972 were submitted by the applicant, as well as similar joint frequency distributions for these same periods with atmospheric stability defined by azimuth angle fluctuations. All joint frequency distributions were submitted in accordance with the requirements of Regulatory Guide 1.23, "Onsite Meteorological Programs," and are acceptable.

We have examined all available onsite meteorological data, and have concluded that the meteorological assumptions selected by the applicant to estimate expected accident dispersion conditions from buildings and vents are adequately conservative; these assumptions were taken from Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized

Water Reactors." Annual average dispersion conditions for releases from buildings and vents have been evaluated using wind speed and direction measurements from the 25-foot level, and with stability defined by vertical temperature gradient between the 25- and 250-foot levels for the period July 1967 to October 1969.

2.3.4 Short-Term (Accident) Diffusion Estimates

In the evaluation of short-term (0-2 hours at the exclusion distance and 0-8 hours at the boundary of the LPZ) accidental releases from buildings and vents, a ground-level release with a building wake factor, cA, of 800 square meters (m^2) was assumed. The relative concentration (X/Q) for 0-2 hours, which is exceeded 5% of the time, was calculated to be 5.3×10^{-4} seconds per cubic meter (sec/m^3) at the exclusion distance of 800 meters (m). This calculation was performed using the model and assumptions described in Regulatory Guide 1.4. This relative concentration is equivalent to dispersion conditions produced by Pasquill Type F stability with a wind speed of 1.0 meter per second (m/sec). The relative concentration, which is exceeded 5% of the time, at the outer boundary of the LPZ (9654 m from the reactors) was calculated to be 2.4×10^{-5} sec/m^3 for the 0-8 hour time period. The corresponding estimated relative concentration at the LPZ for the 8-24 hour time period is 4.8×10^{-6} sec/m^3. For the 1-4 day and 4-30 day time periods, these concentrations are 1.5×10^{-6} and 3.4×10^{-7} sec/m^3, respectively.

2.3.5 Long-Term (Routine) Diffusion Estimates

The highest overland offsite annual average relative concentration for releases from buildings and vents was calculated to be 7.2×10^{-6} sec/m^3 at a location on the minimum site boundary 800 m north of the containment structures.

2.3.6 Conclusions

We have concluded that the applicant's meteorological assumptions taken from Regulatory Guide 1.4 are adequately conservative, and that they provide an acceptable design basis for the calculation of relative concentrations for accidental releases to the atmosphere from buildings and vents.

We will require further discussion with the applicant regarding his proposed program for control room monitoring of meteorological parameters. Issues yet to be resolved include the method to be used for rapid assessment of atmospheric stability, and the control room display of pertinent parameters. The applicant must submit to the staff for evaluation any proposed modification of the present Meteorological Program. No changes to this program may be implemented without prior staff approval.

The applicant must submit at least one additional year of onsite data. These data should preferably be taken from the most recent 2-3 year compilation of onsite meteorological data, and should be submitted in the form of joint frequency distributions of wind speed and direction by atmospheric stability (as defined by vertical temperature gradient) in accordance with

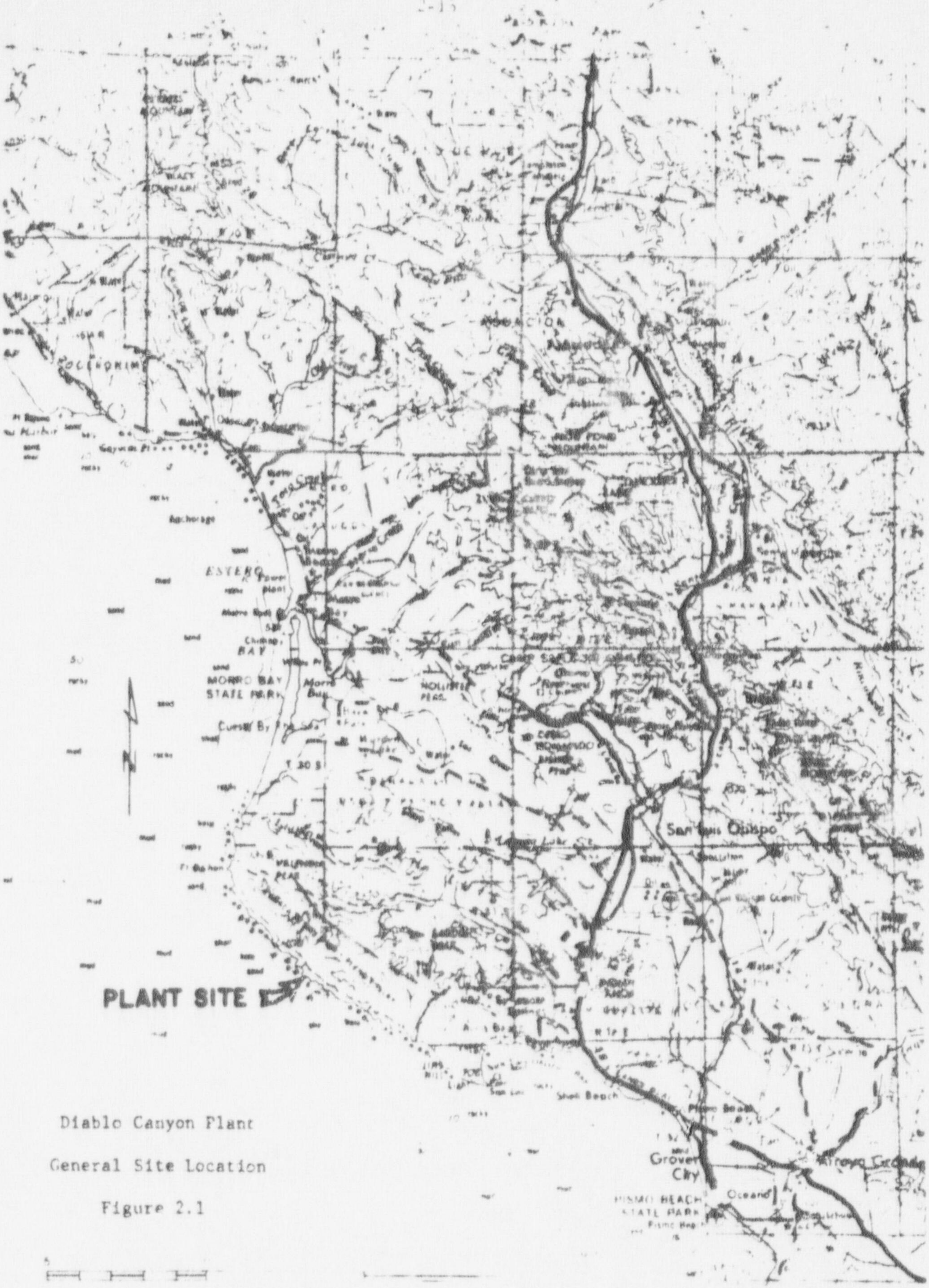
the recommendations of Regulatory Guide 1.23. These data should be submitted in a timely manner in order to permit verification of the relative concentration values.

2.4 Hydrology

Because of delays by the applicant in the submittal of required information, the staff's review of site hydrology has not yet been completed. Our evaluation of this area will be contained in a supplement to this Safety Evaluation Report.

2.5 Geology, Seismology, and Foundation Engineering

Because of delays by the applicant in the submittal of required information, the staff's review of geology, seismology, and foundation engineering has not yet been completed. Our valuation of these areas will be contained in a supplement to this Safety Evaluation Report.



Diablo Canyon Plant

General Site Location

Figure 2.1

Figure 2.2 Location of Site Boundary and Principal Structures

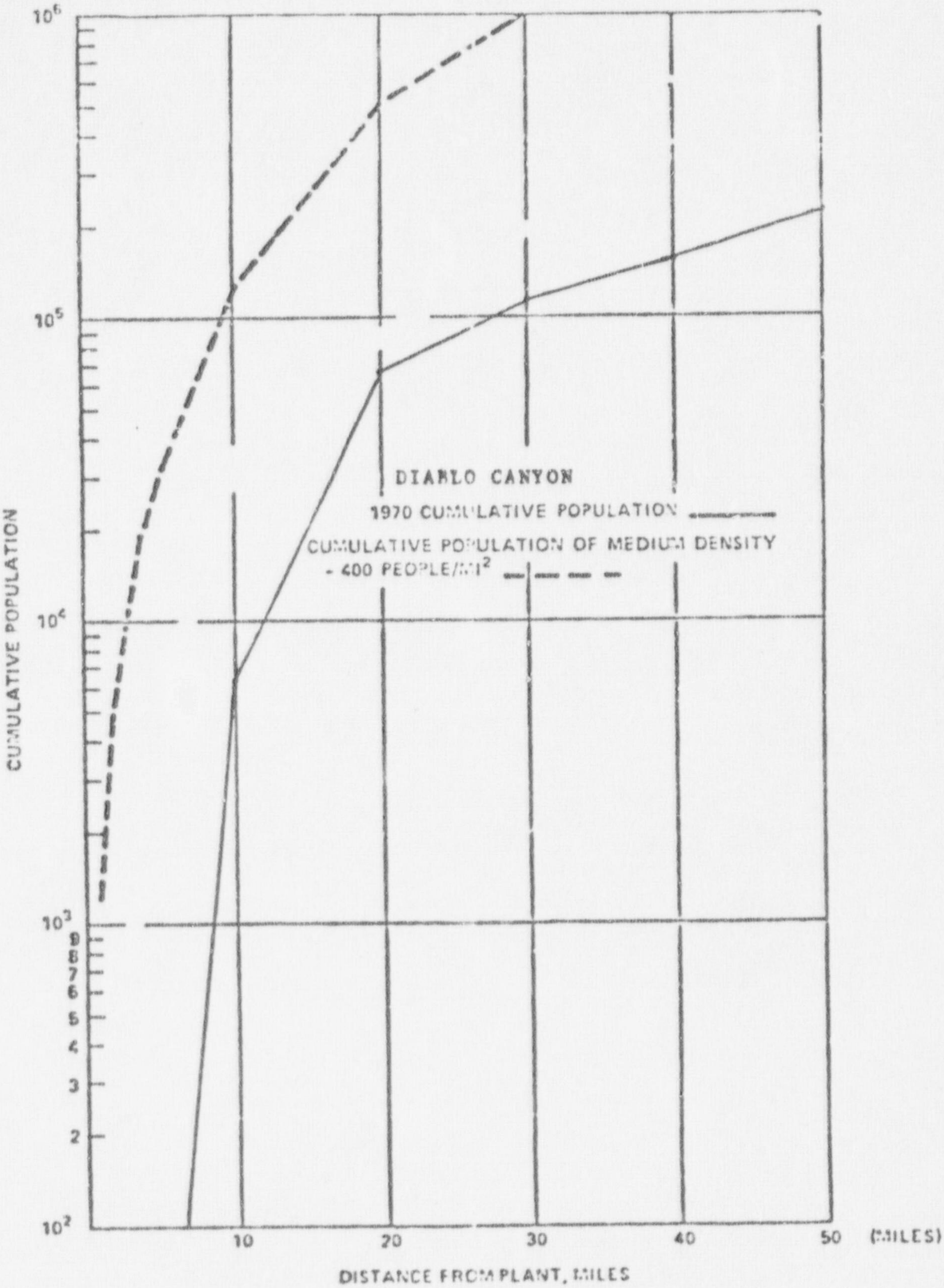


Figure 2.3 CUMULATIVE POPULATION DISTRIBUTION

Bureau of Economic Analysis

Water Resources

Subarea # 1807

Figure 2.4

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 Conformance with AEC General Design Criteria

The Diablo Canyon Units 1 and 2 were designed and are being constructed on the basis of the proposed General Design Criteria (GDC), which were published on July 11, 1967. Since February 20, 1971, when the AEC published the GDC for Nuclear Power Plants, the applicant has attempted to comply with the intent of the newer criteria to the extent that is practical, recognizing previous design commitments. Any exceptions to the 1971 GDC which have been taken because of earlier design or construction commitments are identified in the FSAR in the discussion of the corresponding criterion (see Appendix 3.1A of the FSAR). As a result, our review assessed the plant against the General Design Criteria now in effect, and we have concluded that the plant design conforms to the intent of these newer criteria.

3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

Structures, systems and components important to safety that are required to be designed to withstand the effects of a safe shutdown earthquake (SSE) and remain functional have been properly classified as Seismic Category I (Applicant's Design Class I) items. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary (RCPB), (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the

capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100.

All other structures, systems and components that may be required for operation of the facilities are designed to other than Seismic Category I (Applicant's Design Classes II and III) requirements. Included in this classification are those portions of Category I systems which are not required to perform a safety function. Structures, systems and components important to safety that are designed to withstand the effects of an SSE and remain functional have been identified in an acceptable manner in Table 3.2-4 of the FSAR and on system piping and instrumentation diagrams, Figures 3.2-01 through 3.2-27 of the FSAR.

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for structures, systems and components important to safety with: (1) the Commission's regulations as set forth in AEC General Design Criterion No. 2; (2) the positions set forth in Regulatory Guide 1.29, "Seismic Design Classification"; and (3) industry standards.

We have concluded that structures, systems and components important to safety that are designed to withstand the effects of a safe shutdown earthquake and remain functional have been properly classified as Seismic Category I items in conformance with the Commission's regulations, the applicable Regulatory Guide and industry standards. Design of these

items in accordance with Seismic Category I requirements provides reasonable assurance that the plant will perform in a manner providing adequate safeguards for the health and safety of the public.

3.2.2 System Quality Group Classifications

Fluid system pressure-retaining components important to safety will be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety function to be performed. The applicant has applied a classification system (Code Classes I, II and III) to those fluid containing components which are part of the reactor coolant pressure boundary and other fluid systems important to safety where reliance is placed on these systems, in order to: (1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary; (2) permit shutdown of the reactor and maintain it in the safe shutdown condition; and (3) contain radioactive material. These fluid systems have been classified in an acceptable manner in Tables 3.2-2, 3.2-3 and 3.2-4 and on system piping and instrumentation diagrams, Figures 3.2-01 through 3.2-27 of the FSAR. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for pressure-retaining components such as pressure vessels, heat exchangers, storage tanks, pumps, piping and valves in fluid systems important to safety with: (1) the Commission's Regulations as set forth in AEC General Design

Criterion No. 1; (2) the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50; (3) the positions set forth in Regulatory Guide 1.26, "Quality Group Classifications and Standards"; and (4) industry standards.

We have concluded that fluid system pressure-retaining components important to safety that are designed, fabricated, erected and tested to quality standards in conformance with the Commission's Regulations, the applicable Regulatory Guide, and industry standards are acceptable. Conformance with these requirements provides reasonable assurance that the plant will perform in a manner providing adequate safeguards for the health and safety of the public.

3.3 Wind and Tornado Design Criteria

All Seismic Category I (also referred to as Category I) structures exposed to wind forces have been designed to withstand the effects of the design wind. The design wind specified has a velocity of 80 mph based on a recurrence interval of 100 years. The procedures used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with the International Conference of Building Officials "Uniform Building Code - 1967 Edition."

Although a tornado design criterion was not required as a condition for the granting of construction permits for these units, a review of the tornado resisting capabilities of Category I and certain non-Category I structures has been undertaken by the applicant at the request of the

staff. The objective of the review was to establish capabilities of the Category I structures, as designed and constructed, to withstand tornadic wind pressure and the associated atmospheric pressure drop and tornado borne missile effects. The results of this review indicate that the maximum postulated tornadic wind velocity (300 mph) will not cause a loss of coolant accident or structural damage which would impair containment integrity. All other Category I structures or structures housing Category I components are capable of withstanding the wind effects of at least a 225 mph tornado without failure of major structural elements. Their resistance to the hypothetical missiles corresponds to a 150 mph wind velocity.

The procedures used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings as discussed above. The pressure drop associated with the tornado is treated as a static uniform load applied on vertical and horizontal projected areas of the structures. Tornado missile effects have been determined using procedures which are discussed in Section 3.5 of this report. The total effect of the tornado on Category I structures was determined by the appropriate combination of the individual effects of the tornado wind pressure, pressure drop and associated missiles. Structures are arranged on the plant site and protected in such manner that the collapse of structures which do not have tornado resisting capability will not affect those which must withstand the tornado effects.

The criteria used in the design of Category I structures to account for the loadings due to specific winds postulated to occur at the site, and the methods used in determining those loads provide a conservative basis for the plant design. The use of these loading criteria provides reasonable assurance that, in the event of wind (or a tornado), the structural integrity and safety function of Seismic Category I structures will not be impaired by the specified environmental forces.

We have concluded that conformance with these criteria is an acceptable basis for satisfying the requirements of AEC General Design Criterion No. 2.

3.4 Water Level (Flood) Design Criteria

The design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level of the site is discussed in Section 2.4 of the FSAR (Hydrology). This discussion indicates that it is not possible to develop sufficient ponding to flood safety related buildings. Thus, the depth of water at the plant location for the Probable Maximum Flood is zero.

The auxiliary saltwater system is the only safety related system that has components that would be affected by the design combination of tsunami-storm wave activity. The intake structure is designed so that the auxiliary saltwater pumps are housed in separate water-tight compartments, which assures that the pumps will operate during the storm. The auxiliary saltwater system intake structure is also provided with a ventilation system that is above the postulated flood level.

On the basis of our review, we have concluded that the design of the intake structure to withstand the effects of tsunami-storm wave activity provides reasonable assurance that the safety related equipment is adequately protected and may be expected to perform its required safety functions. Furthermore, we conclude that the design satisfies the requirements of AEC General Design Criteria Nos. 2 and 4 as related to the environmental design bases for safety related structures.

3.5 Missile Protection Criteria

The design of essential structures and vital equipment has taken into account the effects of a spectrum of tornado-borne missiles and internally generated missiles associated with component overspeed failures and missiles that could originate from high-pressure system ruptures. The design assures that there will be no loss of function of a Seismic Category I structure or of essential system or component functions as a result of missiles. The missiles applicable to each of these structures and pieces of equipment have been adequately identified, and their characteristics defined.

Initially, the tornado resisting capability of all Seismic Category I structures was calculated for a limited spectrum of tornado-borne missiles (see Page 3.3-5 of the FSAR). In response to our request, the applicant increased this missile spectrum to include the following items that could be present at the site or dislodged from structures by tornadic winds to become missiles: (1) a utility pole 13.5 inches in

diameter and 35 feet long, with a density of 43 pounds per cubic foot (lb/ft^3); (2) a one-inch diameter solid steel rod, 3 feet long with a density of 490 lb/ft^3; and (3) pieces of 6 and 12 inch schedule 40 pipe, each 15 feet long with a density of 490 lb/ft^3. In general, essential components contained in Seismic Category I structures are inherently protected by virtue of the fact that the seismic and other design requirements result in structures that are tornado resistant.

The analysis of structures, shields and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structure. In the second step of the analysis, the overall structural response of the target when impacted by a missile was determined using established methods of impactive analysis.

The design procedures used to determine the effects and loading on Seismic Category I structures by design basis missiles selected for the plant provide a conservative basis for engineering design to assure adequate protection from the effects of missile impacts. The use of this information provides reasonable assurance that, in the event of design basis missiles striking Seismic Category I structures, the structural integrity of structures will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic

Category I systems and components located within these structures are, therefore, expected to be adequately protected against the effects of missiles. Conformance with these missile protection design procedures is an acceptable basis for satisfying the requirements of AEC General Design Criterion No. 4.

Based on the results obtained from the revised tornado missile analysis, the applicant evaluated the plant layout to determine what systems or components necessary for safe reactor shutdown could be damaged by potential tornado missiles. As a result of this evaluation, the applicant has proposed design modifications to further improve the post-tornado availability of components in the emergency power system. These modifications consist of tornado missile barriers that protect the louvers in the diesel generator rooms. We find these modifications to be acceptable. With the exception of these modifications, the applicant's analysis indicated that existing shielding or equipment separation provided adequate protection to assure that safe shutdown of the reactor can be achieved and maintained. We have reviewed the information submitted by the applicant and concur with his conclusions.

3.6 Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping

The applicant's criteria which were used for identifying high energy fluid piping and for postulating pipe break locations, break orientations and break flow areas are equivalent to the criteria for piping inside

containment set forth in Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," and are also consistent with the criteria stated in the A. Giambusso letter of December 18, 1972 for piping outside of containment. The provisions for protection against the dynamic effects associated with pipe rupture (inside containment) and the resulting discharging coolant provide acceptable assurance that, in the event of the occurrence of the combined loadings imposed by an earthquake of the magnitude specified for the safe shutdown earthquake (SSE) and a concurrent single pipe break of the largest pipe at any one of the design basis break locations, the following conditions and safety functions will be accommodated and assured:

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping;

(2) The reactor emergency core cooling systems can be expected to perform their intended function; and

(3) Structures, systems and components important to safety will be appropriately protected.

The analytical methods and procedures that were used to determine pipe motion subsequent to rupture and the pipe-whip restraint dynamic interaction appropriately consider the structural characteristics of the system. The pipe-whip restraints are designed to withstand the resultant loadings in accordance with acceptable criteria.

On the basis of our review, we have concluded that the criteria used for the identification, design and analysis of piping systems where

postulated breaks may occur constitute an accep.able design basis in meeting the applicable requirements of AEC General Design Criteria Nos. 1, 2, 4, 14, & 15, and are consistent with Regulatory staff positions for plants currently under review for operating licenses.

With regard to piping outside the containment, the applicant has submitted a final report which presents the results of the investigations conducted to determine the consequences of postulated ruptures of high energy fluid piping outside the containment. Included in this report are definitions of criteria and methods employed in the analyses, the identification of high energy fluid piping outside containment, and the structures and systems required for safe shutdown of the reactor following postulated ruptures of this piping. A summary of the analysis results for breaks in the main steam piping between the containment and the turbine stop valves, and breaks in the feedwater piping between the containment and the feedwater pumps, including proposed design modifications, has been reviewed. Pipe break effects analyzed included jet impingement, pressurization of compartments, water flooding, and the environmental effects of pressure, temperature, and humidity.

We have reviewed the material presented and have found it to be in accord with our requirements. For the breaks in the main steam and feedwater piping that were analyzed, we conclude that the applicant has developed an adequate design so that postulated pipe breaks outside of containment will not prevent safe shutdown of the reactor. The results

of our final evaluation of the applicant's report on pipe break outside containment will be documented in a supplement to this Safety Evaluation Report.

3.7 Seismic Design

The seismic design response spectra curves were presented in the PSAR and approved prior to the issuance of the construction permits for Units 1 and 2 of the Diablo Canyon Plant. The modified earthquake time histories used for component equipment design are adjusted in amplitude and frequency to envelope the response spectra specified for the site. We conclude that the seismic input criteria proposed by the applicant provide an acceptable basis for seismic design.

Modal response spectrum, multi-degree-of-freedom, and normal mode-time history methods are used for the analysis of all Seismic Category I structures, systems and components. The vibratory motions and the associated mathematical models account for the soil structure interaction and the coupling of all Category I structures and plant equipment. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the modal response spectrum method was used. The absolute sum of responses is used for closely spaced frequencies. Horizontal and vertical floor spectra inputs used for design and test verification of structures, systems and components were generated by the normal mode-time history method. Torsional loads have been adequately accounted for in the seismic analysis

of the Category I structures. Vertical ground accelerations were assumed to be 2/3 of the horizontal ground accelerations, and the horizontal and vertical effects were combined simultaneously. Constant vertical load factors were employed only where analysis showed sufficient vertical rigidity to preclude significant vertical amplifications in the seismic system being analyzed.

We have reviewed the FSAR and applicable Amendments and find the seismic system and subsystem dynamic analysis methods and procedures proposed by the applicant to be acceptable.

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure corresponds to the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes." Supporting instrumentation will be installed on Category I structures, systems, and components in order to provide data for the verification of the seismic responses determined analytically for such Seismic Category I items.

We conclude that the seismic instrumentation program proposed by the applicant is acceptable.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

The reactor coolant system is enclosed in a reinforced concrete containment as described in Section 3.8.2 of the FSAR. The containment

structure has been designed in accordance with applicable subsections of the ASME Boiler and Pressure Vessel Code, Section III, and ACI 318 to resist various combinations of dead loads, live loads, environmental loads (including those due to wind, the OBE, and the SSE), and loads generated by the design basis accident (including pressure, temperature and associated pipe rupture effects). The effects associated with postulated rupture of high-energy pipes such as reaction and jet impingement forces and impact effects of whipping pipes have also been considered in the design.

The static analysis for the containment shell and base utilizes methods that have been previously applied. Likewise, the liner design for the containment employs methods similar to those previously accepted. The choice of the materials, the arrangement of the anchors, the design criteria and design methods are similar to those evaluated for previously licensed plants. Materials, construction methods, quality assurance and quality control measures are covered in the FSAR, and, in general, are similar to those used for previously accepted facilities.

Prior to operation, the containment structure will be subjected to an acceptance test in accordance with the recommendations given in Regulatory Guide 1.18, "Structural Acceptance Test for Concrete Primary Reactor Containments." During this test the internal pressure will be 1.15 times the containment design pressure of 47 psig.

Although the containment is considered to be a prototype, the applicant took exception to paragraph C5 of Regulatory Guide 1.18 in that the

strain measurements will be taken near inside and outside faces of the containment walls only, omitting the midpoint location. We find this to be acceptable for an operating license.

The criteria used in the analysis, design and construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff. The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural design criteria; the materials, quality control and special construction techniques; and the testing requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring inside and outside the containment, the structure may be expected to withstand the specified design conditions without impairment of its structural integrity and safety function. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria Nos. 2, 4, 16, and 50.

3.8.2 Concrete and Structural Steel Internal Structures

The containment interior structure consists of a shield wall around the reactor, secondary shield walls, and other interior walls, compartments

and floors. The principal code used in the design of concrete internal structures is ACI 318-63, "Building Code Requirements for Reinforced Concrete." For steel internal structures the AISC document, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," was used. The containment concrete and steel internal structures have been designed to resist various combinations of dead and live loads, accident induced loads (including pressure and jet loads), and seismic loads. The load combinations used cover those cases likely to occur and include all loads which may act simultaneously. The design and analysis procedures that have been used for the internal structures are identical with those approved for previously licensed plants and, in general, are in accordance with procedures delineated in the ACI 318-63 Code and in the AISC Specification for concrete and steel structures, respectively.

The containment internal structures have been designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits are, in general, based on the ACI 318-63 Code and on the AISC Specification, modified as appropriate for the specific load combinations. The materials of construction, and their fabrication, construction and installation, are also in accordance with the previously mentioned codes and specifications.

The criteria that have been used in the analysis, design and construction of the internal structures of the containment, to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in accordance with established criteria, codes, standards, and specifications acceptable to the Reg ory staff. The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; and the materials, quality control and special construction techniques; provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring inside the containment, the interior structures may be expected to withstand the specified design conditions without impairment of their structural integrity and safety function. Conformance with these criteria, codes, specifications and standards constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria Nos. 2 and 4.

3.8.3 Other Seismic Category I Structures

Seismic Category I (also referred to as Category I) structures other than containment and its interior have been built from structural steel and reinforced concrete members. The structural components consist of slabs, walls, beams and columns. The design method for reinforced concrete followed that specified in the ACI 318-63 Code. Structural steel components were designed in accordance with the AISC specifications. The concrete and steel Category I structures have been designed to resist

various combinations of dead loads, live loads, environmental loads (including those due to wind, the OBE and the SSE), and loads generated by postulated ruptures of high energy pipes (such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes). The design and analysis procedures that have been used for these Category I structures are the same as those approved on previously licensed plants. The various Category I structures have been designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits have been modified as appropriate for load combinations that are considered extreme. The materials of construction, and their fabrication, construction and installation are in accordance with previously mentioned codes and specifications, i.e., ACI 318-63 and AISC specifications.

In order to provide assurance that the function of Category I equipment located in the turbine building and the intake structure (both Seismic Category II structures) will not be adversely affected in the event of a safe shutdown earthquake, these structures have been reviewed for that earthquake to assure that they would not collapse. These analyses have shown that some yielding would occur in the turbine building, but that this yielding would be limited to safe values. All stresses in the intake structure would be less than yield. For components located in both of these structures, the occurrence of an SSE will not impair the capability of Seismic Category I equipment to perform its safety related function.

The criteria used in the analysis, design and construction of all the plant Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff. The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control and special construction techniques; provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within these structures, they may be expected to withstand the specified design conditions without impairment of their structural integrity and safety function. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria Nos. 2 and 4.

3.8.4 Foundations and Concrete Supports

Foundations of Category I structures are described in Sections 3.8.1 and 3.8.2 of the FSAR. Primarily, these foundations are reinforced concrete of the mat type. The major code used in the design of these concrete mat foundations is ACI 318-63. These concrete foundations have

been designed to resist various combinations of dead loads, live loads, environmental loads (including those due to wind, the OBE and the SSE), and loads generated by postulated ruptures of high energy pipes. The design and analysis procedures that have been used for these Category I foundations are the same as those approved on previously licensed plants and, in general, are in accordance with procedures delineated in the ACI 318-63 Code. The various Category I foundations have been designed and proportioned to remain within limits established by the Regulatory staff under the various load combinations. These limits have been modified as appropriate for load combinations that are considered extreme. The materials of construction, and their fabrication, construction and installation, are in accordance with the ACI 318-63 Code.

The criteria used in the analysis, design and construction of plant Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff. The use of these criteria as defined by applicable codes, standards and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; and the materials, quality control and special construction techniques; provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various

postulated accidents, the foundations may be expected to withstand the specified design conditions without impairment of their structural integrity and safety function. Conformance with these criteria, codes, specifications, and standards constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria Nos. 2 and 4.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

In order to assure that the vibration of piping systems is within acceptable levels, the applicant will conduct a piping vibration operational test program. The preoperational vibration dynamic effects test program that will be conducted on safety related ASME Class A and Class B piping systems and piping restraints during startup and the initial operating conditions is acceptable to the staff. These tests will provide adequate assurance that the piping and piping restraints of the systems have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation, and are consistent with recent Regulatory staff positions concerning preoperational piping dynamic effects test programs for other plants. Compliance with this test program constitutes an acceptable basis for partial fulfillment of the requirements of AEC General Design Criterion No. 2.

With regard to flow-induced vibrational testing of reactor internals for Diablo Canyon Units 1 and 2, the applicant has identified Indian

Point Unit 2 as the prototype plant for Unit 1, and has established Trojan as the prototype plant for design verification of Diablo Canyon Unit 2. Two prototypes have been designated because Unit 1 has a thermal shield while Unit 2 utilizes neutron pads. These designations have been made in accordance with the provisions of Regulatory Guide 1.20, "Vibration Measurements on Reactor Internals." The applicant will perform additional confirmatory vibration testing and subsequent visual inspection as part of the Diablo Canyon preoperational tests to provide added confirmation of the capability of the structural elements of the reactor internals to sustain flow-induced vibrations. The proposed program is consistent with the positions stated in Regulatory Guide 1.20.

We have reviewed the preoperational vibration test program proposed by the applicant for verifying the design adequacy of the reactor internals under loading conditions that will be comparable to those experienced during operation. We find this program to be acceptable. The combination of tests, predictive analysis, and post-test inspection will provide adequate assurance that the reactor internals can be expected, during their service lifetime, to withstand the flow-induced vibrations resulting from reactor operation, without loss of structural integrity. We have reviewed the preoperational vibration test program that will be performed in accordance with Regulatory Guide 1.20 for assurance that

it constitutes an acceptable basis for demonstrating the design adequacy of the reactor internals in satisfying the applicable requirements of AEC General Design Criteria Nos. 2 and 14. We find that the program does constitute such an acceptable basis. If the test results on the prototype reactors mentioned previously are not found to be acceptable, we will require that the test programs for Diablo Canyon Units 1 and 2 be expanded to obtain the appropriate test data.

The applicant has performed a dynamic system analysis of the reactor internals and of the broken and unbroken piping loops. This analysis provides an acceptable basis for confirming the structural design adequacy of the reactor internals and the unbroken piping loops to withstand the combined dynamic effects of the postulated occurrence of a design basis accident (LOCA) and a safe shutdown earthquake. The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections or displacements of any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling is impaired. The assurance of structural integrity of the reactor internals under the postulated SSE and the most severe LOCA conditions provides added confidence that the design can be expected to withstand a spectrum of lesser pipe breaks and seismic loading combinations.

We have concluded that the applicant's design procedures and analytical techniques represent acceptable bases for structural design of the reactor internals for Diablo Canyon Units 1 and 2.

3.9.2 ASME Code Class 2 and 3 Components

All Category I pressure retaining systems, components and equipment outside of the reactor coolant pressure boundary (RCPB) are designed to sustain normal loads, anticipated transients, the operating basis earthquake, and the safe shutdown earthquake within design limits which are comparable to those outlined in AEC Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid Systems Components." The specified design basis combinctions of loadings as applied to the design of the safety related ASME Code Class 2 and 3 pressure-retaining components in systems classified as Seismic Category I provide reasonable assurance that in the event (1) an earthquake should occur at the site, and (2) other upset, emergency or faulted plant transients should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of the system components to withstand the most adverse combinations of loading events without gross loss of structural integrity. The applicant's design load combinations and

associated stress and deformation limits specified for all ASME Code Class 2 and 3 components constitute an acceptable basis for design in satisfying AEC General Design Criteria Nos. 1, 2, and 4, and are consistent with recent Regulatory staff positions.

The applicant has conducted component test programs, supplemented by analytical predictive methods, which provide adequate assurance that the capability of ASME Code Class 2 and 3 active pumps and valves (1) to withstand the imposed loads associated with normal, upset, emergency, and faulted plant conditions without loss of structural integrity, and (2) to perform the "active" function (i.e., valve closure or opening), is confirmed under conditions and combinations of conditions comparable to those expected when a safe plant shutdown is to be effected, or the consequences of an accident are to be mitigated.

We have concluded that the design and analytical procedures used by the applicant provide reasonable assurance of pump and valve operability.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Instrumentation and electrical components required to perform a safety function are designed to meet Seismic Category I requirements. These requirements established by the seismic system analysis have been incorporated into equipment specifications to assure that the equipment purchased or designed meets seismic requirements equal to or in excess of the requirements for Category I components; this assurance can be provided either by appropriate analysis or by qualification testing.

The applicant has implemented a seismic qualification program for Category I instrumentation and electrical equipment and the associated supports for that equipment to provide assurance that such equipment can be expected to function properly, and that structural integrity of the supports will not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and the conditions of post-accident operation. The applicant has based his instrumentation and equipment qualification test program on Topical Report WCAP-8021, "Seismic Testing of Electrical and Control Equipment (PG&E Plants)," and has also referenced IEEE Std 344-1971. This WCAP report is presently under review by the staff (see Section 7.8 of this report). This qualification program, when completed, will constitute an acceptable basis for satisfying staff requirements and AEC General Design Criterion No. 2. Resolution of this item will be reported on in a supplement to this Safety Evaluation Report.

4.0 REACTOR

4.1 Summary Description

The Nuclear Steam Supply System design for Diablo Canyon is similar to that reviewed and approved for the Zion Nuclear Power Station Units 1 and 2 and D. C. Cook Nuclear Plant Units 1 and 2 except for the following:

(1) Diablo Canyon Units 1 and 2 have initial core power levels of 3338 and 3411 MWt, respectively, which are 2.7 and 5% higher than the Zion and D. C. Cook ratings of 3250 MWt. The power levels for the Diablo Canyon facility apply to either the 15x15 or the 17x17 fuel assemblies, as discussed in Section 4.4 of this report;

(2) Diablo Canyon will use a 17x17 fuel assembly design as compared to the 15x15 design for the Zion and D. C. Cook facilities;

(3) Diablo Canyon has eliminated the loop isolation valves which were included in the Zion reactor coolant system design;

(4) The design core inlet reactor coolant temperatures for Diablo Canyon Units 1 and 2 are 544.4 and 545°F, respectively, as compared to the Zion Unit 2 and D. C. Cook values of 530.2 and 536.3°F, respectively.

4.2 Mechanical Design

4.2.1 Fuel

The Diablo Canyon fuel assembly consists of 264 fueled rods, 24 guide thimbles, and 1 instrumentation thimble plus ancillary hardware arranged in a 17x17 array. The instrumentation thimble is at the center of the

assembly and facilitates the insertion of neutron detectors. The guide thimbles provide channels for inserting various reactivity controls. The fuel rods contain enriched uranium dioxide hermetically clad in Zircaloy-4. The assembly is supported at both ends by stainless steel nozzles. Alignment and transverse spacings are maintained by 8 spacer grids equally spaced along the axis of the fuel assembly.

The Diablo Canyon fuel assembly (17x17) is mechanically similar to the 15x15 Westinghouse assemblies used previously in the D. C. Cook and Zion reactors. Those mechanical aspects which are different are exhibited in Table 4.1 where the comparison is made with the D. C. Cook reactors. The differences are essentially geometric, resulting in a lower linear power density and other increased safety margins, as discussed later.

The evaluation of the Westinghouse fuel mechanical design is based upon mechanical tests, in-reactor operating experience and engineering analyses. Additionally, the in-reactor performance of the design will be subject to the continuing surveillance programs to be conducted by Westinghouse and the individual utilities. These programs continually provide confirmatory and current design performance information.

In our evaluation of the fuel thermal performance we assume that densification of uranium dioxide fuel pellets may occur during irradiation in power reactors. The initial density of the fuel pellets, and the size, shape and distribution of pores within the fuel pellet influence the densification phenomenon. The effects of densification

on the fuel rod will increase the stored energy, increase the linear thermal output, increase the probability for local power spikes, and decrease the thermal conductance. The primary effects of densification on the fuel rod mechanical design are manifested in calculations of time-to-collapse of the cladding and fuel-cladding gap conductance. Time-to-collapse calculations predict the time required for unsupported cladding to become dimensionally unstable and to flatten into an axial gap caused by fuel pellet densification. Gap conductance calculations predict the decrease in thermal conductance due to opening of the fuel-clad radial gap.

The engineering methods used by Westinghouse to analyze the fuel thermal performance have been previously submitted in the Westinghouse Topical Report WCAP-8218, "Fuel Densification, Experimental Results and Model for Reactor Application," dated October 1973, and have been reviewed by the Regulatory staff. The results of our review were reported in "Technical Report on Densification of Westinghouse PWR Fuel," issued by the Commission on May 4, 1974. On the basis of our review we have concluded that the applicant has considered the effects of densification on the Diablo Canyon fuel assemblies in a manner which adequately describes the fuel behavior.

All fuel rods will be internally prepressurized with helium during final welding to minimize cladding compressive stresses during service. The level of prepressurization is designed to preclude any cladding tensile stresses throughout operation due to total internal pressure.

Substantially all of the in-reactor operating experience with Westinghouse fuel rods and 15x15 assemblies is applicable to the Diablo Canyon fuel since the 17x17 fuel assembly is a slight mechanical extrapolation from the 15x15 assembly. The range in design parameters for which this in-reactor experience is applicable is tabulated in Table 4.2. The assemblies referred to in this Table have been irradiated up to 6 years and have had peak exposures of 30,000 megawatt-days/metric ton (MWD/MT), totaling more than 70 million megawatt hours of power generation.

During this power reactor service a small fraction of the fuel rods have experienced defects. There has been no instance where cladding defects have threatened either the plant or the public safety. Cladding defects were caused by either excessive manufacturing impurities, excessive coolant cross-flow velocities and/or fuel pellet densification. These causes have been amended by modifications to both the manufacturing procedures and the plant coolant system. The fuel related modifications required readjustments of the magnitude of a design characteristic rather than a redesign of the fuel assembly. Fuel rods and assemblies identical to the Diablo Canyon design have not yet experienced power reactor service. However, the current use of similar fuel rods and assemblies has yielded operating experience that provides confidence in the acceptable performance of the 17x17 fuel rods and assemblies.

Out-of-reactor mechanical tests have been performed on typical 15x15 fuel assemblies. These tests demonstrate acceptable mechanical performance of the 15x15 fuel assembly. Since the 17x17 assembly is

a slight mechanical extrapolation from the 15 x 15 assembly, we expect the mechanical behavior of the two assemblies to be similar; therefore, we have concluded that the 17 x 17 fuel assembly is acceptable.

Verification tests on the 17 x 17 assemblies have been completed and reported in WCAP-8278, "Hydraulic Flow Test of the 17 x 17 Fuel Assembly;" and WCAP-8236 "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident." The staff has reviewed these two topical reports and concluded that they are acceptable. The grid tests were made on assemblies with 7 grids; Westinghouse will document in a topical report the justification for applying the test results to 8 grid assemblies. Also, the first phase of the single rod burst tests has been completed and will also be documented. We will review the documentation and will report the results of our evaluation in a supplement to this Safety Evaluation Report prior to a decision concerning the issuance of operating licenses for Diablo Canyon Units 1 and 2.

Performance of the fuel during operation will be indirectly monitored by measurement of the activity of both the primary and the secondary coolant for compliance with technical specification limits. The first available irradiated 17 x 17 fuel assemblies and rods will undergo an extended surveillance program following each cycle of operation. Onsite examinations will include fuel rod integrity, fuel rod and fuel assembly dimensions and alignment, and surface deposits. Details of the surveillance program will be reported on in a supplement to this Safety Evaluation Report.

In addition, Region III of the Diablo Canyon core will have one removable fuel rod assembly conceptually similar to removable rod assemblies used in Zion-1, Surry-1, Point Beach, San Onofre and others. Eighty-eight (88) fuel rods will be removable to facilitate interim and end of life (EOL) fuel evaluation as a function of exposure. The removable fuel rods are identical to the other fuel rods in the core and will provide direct inspection results on fuel which performed under the combined thermal, hydraulic and nuclear conditions expected during normal operation. The technical specification requirements will result in surveillance of these 17 x 17 irradiated fuel rods which have been precharacterized.

We have concluded, subject to confirmation of the previously cited required documentation, that based on (1) operating experience with similar fuel, (2) the results of out-of-reactor tests on an assembly of similar design, (3) the increased thermal margins which the 17 x 17 fuel provides, (4) the technical specification requirements to monitor and limit off-gas and effluent activity, and (5) the existence of a continuing fuel rod surveillance program which includes destructive and non-destructive post irradiation examinations, the cladding integrity of the 17 x 17 fuel will be maintained and significant amounts of radioactivity will not be released. Furthermore, on the basis of our review of the Westinghouse Topical Report WCAP-8236, we have also concluded that neither accidents or earthquake induced loads will result in either an inability to cool the fuel or interference with control rod insertion.

4.2.2 Reactor Vessel Internals

The design and vibrational test programs for the reactor vessel internals are discussed in Section 3.9.1 of this report.

We have reviewed the selection of materials for the reactor vessel internals required for distributing the coolant flow to achieve acceptable heat transfer performance for all modes of reactor operation. The major containment and support member of the reactor internals is the lower core support structure. For the Diablo Canyon Plant, the full thermal shield in Unit 1 was replaced on Unit 2 by a neutron panel assembly. The principal material for the reactor internals is Type 304 stainless steel. Type 316 stainless steel and Inconel are specified for small parts such as bolts, dowel pins, and inserts. Type 403 stainless steel is specified for parts requiring a yield strength above 90,000 psi.

All materials used are compatible with the reactor coolant, and have performed satisfactorily in similar applications. They meet the ASME Boiler and Pressure Vessel Code requirements (1973). Undue susceptibility to intergranular stress corrosion cracking will be prevented by avoiding the use of sensitized stainless steel as recommended in Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel."

The use of materials proven to be satisfactory by actual service experience, and avoidance of sensitization by the methods recommended in Regulatory Guide 1.44 will provide reasonable assurance that the reactor vessel internals will not be susceptible to failure by corrosion or stress corrosion cracking

4.3 Nuclear Design

4.3.1 General

Diablo Canyon Units 1 and 2 are similar to several Westinghouse designed 4 loop reactors that will use the 17 x 17 fuel design and that are currently being reviewed for operating licenses. The staff has recently completed a generic review of the Westinghouse 17 x 17 fuel design (described in WCAP-8185, "Reference Core Report 17 x 17"), and has concluded that the nuclear design is acceptable. We have also concluded that the nuclear design applies directly to the Diablo Canyon reactors, and is therefore acceptable. The nuclear characteristics of the 17 x 17 fuel assemblies are essentially the same as those of the previous 15 x 15 Westinghouse design. As a result, there are no changes in control requirements, control rod patterns and reactivity worths, and xenon stability. The analytical methods employed in the design of this core are the same as those used for Westinghouse reactors in recent years, and are acceptable.

The staff has performed certain independent analyses to support its conclusions of acceptability. The results of these analyses are discussed in the following sections.

4.3.2 Power Distribution

The primary feature of the 17 x 17 fuel design relating to physics and power distribution monitoring considerations is the increase in the number of linear feet of fuel provided. For a given core power rating and peaking factor (F_Q), this leads to a reduction in the average and peak linear

power densities (LPD). Thus, the average LPD for Diablo Canyon Unit 2 decreased from 7.03 kW/ft to 5.44 kW/ft with the change from 15 x 15 to 17 x 17 fuel. At 102 percent of full power, the peak LPD associated with an F_Q of 2.32 is now 12.9 kW/ft; by way of comparison, the LPD which would produce a 2200°F clad temperature in a LOCA analysis performed in accordance with the Final Acceptance Criteria is 15.7 kW/ft. The Unit 1 average LPD is 5.33 kW/ft which is correspondingly more conservative.

The applicant has proposed to take credit for correct normal operator action in determining the peaking factor used to define initial conditions for accident analyses. The information presented indicates F_Q would be limited to 2.32 (including fuel densification power spikes), whereas with the previously used axial offset relation, the F_Q limit would be about 2.5. The applicant's plan would eliminate most of the xenon transient effects on F_Q that otherwise would occur in load following.

.3.3 Reactivity Coefficients

The staff has made an independent comparison of the beginning of life (BOL) moderator and Doppler reactivity coefficients for the 15 x 15 and 17 x 17 fuel assemblies using methods equivalent to those employed by the applicant. The calculated BOL isothermal moderator temperature coefficient in the operating temperature range is 0.1 to 0.2 x 10^{-4}/°F more negative with the 17 x 17 fuel assembly than with the 15 x 15 design. The moderator temperature coefficient varies from about 0 to -3.5 x 10^{-4}/°F over the first cycle. The calculated BOL isothermal Doppler coefficient in the operating temperature range is approximately 2 percent more negative

with the 17 x 17 fuel assembly that with the 15 x 15 design. These effects are due to the slightly higher resonance absorption in U-238 in the 17 x 17 fuel assembly lattice.

The reactivity coefficients of cores using the 15 x 15 fuel assembly have been determined in recent reactor startup test programs, e.g., Surry Units 1 and 2, and compare favorably with predictions.

Moderator temperature coefficients as a function of temperature and soluble boron concentration shown in Figure 4.3-30 of the FSAR for the 17 x 17 fuel assembly core are almost identical to those previously reported for the 15 x 15 design. The Doppler coefficient for the 17 x 17 assembly core in Figure 4.3-27 of the FSAR contains the x-y spatial power shape weighting and is not directly comparable to the pointwise Doppler coefficient reported for the 15 x 15 fuel assembly (the pointwise value agrees well with the staff's calculation). However, the x-y power shape weighting factor appears reasonable, and this form of the data is appropriate for use in the study of axial power shapes using the PANDA code. We conclude that the reactivity coefficients are acceptable for use in control and safety analyses related to the 17 x 17 fuel assembly design.

4.4 Thermal and Hydraulic Design

The reactors for Diablo Canyon Units 1 and 2 are designed to operate at core power levels of 3338 and 3411 MWt, respectively, which correspond to net electrical outputs of 1084 and 1106 MWe. The thermal and hydraulic

design has been evaluated on the basis of 3411 MWt. Although the Diablo Canyon Plant will utilize a 17 x 17 fuel assembly, whereas the original Diablo Canyon submittal incorporated a 15 x 15 fuel assembly, the following thermal and hydraulic parameters have remained unchanged:

	Unit 1		Unit 2
(1)	Core power	(1)	Core power
(2)	System pressure	(2)	Vessel loop flow rate
(3)	Coolant inlet temperature	(3)	System pressure
(4)	Open lattice fuel rod array.	(4)	Coolant inlet temperature
		(5)	Core and vessel average and exit coolant temperatures
		(6)	Open lattice fuel rod array.

The basic modification to the Diablo Canyon core design is an increase in the number of fuel rods and the resulting reduction in peak and average linear heat generation rates and heat flux.

The principal criterion for the thermal and hydraulic design of a reactor is to prevent fuel rod damage by providing adequate heat transfer for the various core heat generation patterns occurring during normal operation, operational transients, and transient conditions resulting from faults of moderate frequency. The fuel damage limits and thermal-hydraulic criteria used to evaluate the fuel performance of the Diablo Canyon reactors are the same for the 17 x 17 design as for the previously

proposed 15 x 15 design. These damage limits for normal operation, operational transients, and any transient conditions arising from faults of moderate frequency are (1) departure from nuclear boiling will not occur on at least 95 percent of the limiting fuel rods at a 95 percent confidence level (95/95 criterion); (2) the maximum fuel temperature shall be less than the melting temperature of UO_2; (3) at least 95.5 percent of the thermal flow will pass through the fuel rod region of the core; and (4) the permitted modes of operation shall not lead to hydrodynamic instability. In order to show compliance with these criteria, the applicant performed DNB and fuel temperature calculations as well as flow distribution and flow instability analyses. Some of the important thermal and hydraulic parameters of the 17 x 17 design and the results of the DNB calculations are presented in Table 4.3. For comparison purposes, corresponding information representative of the former Diablo Canyon 15 x 15 and Zion designs has also been included in Table 4.3.

The Diablo Canyon 17 x 17 design has approximately the same DNB margin as the 15 x 15 design, along with an increased fuel temperature margin. It should be noted that the effects of fuel densification were included in the 17 x 17 calculations, while these effects were missing from the 15 x 15 calculations. Also, the Diablo Canyon 17 x 17 DNB calculations included approximately a 14 percent margin on DNB for the following reasons: (1) to allow incorporation of the final results of the DNB and mixing tests;

(2) to allow incorporation of the final results of the hydraulic tests (D-loop tests); and (3) to allow for any fabrication tolerances larger than those presently used.

The first part of the DNB tests, utilizing uniformly heated rods, was completed and reported in WCAP-8296, "Effect of 17 x 17 Fuel Assembly Geometry on DNB." The results indicate the following: (1) the previously used DNB correlation (W-3 correlation with a modified space factor) must be multiplied by 0.88 in order to show agreement with the 17 x 17 data; (2) the use of a thermal diffusion coefficient (TDC) of 0.038 is conservative; and (3) a DNBR value of 1.275 corresponds to the 95/95 criterion. Since only data with uniformly heated rods were considered, it is uncertain at the present time whether further adjustments in the correlation or in the DNBR corresponding to the 95/95 criterion are needed to cover the expected range in axial power shapes. Additional DNB tests with non-uniform axial heating are planned for December, 1974. The results of these tests, together with those reported in WCAP-8296, must be used to set technical specification limits for the Diablo Canyon 17 x 17 fuel assembly design.

Although Westinghouse does not expect further changes in the DNB correlation or in the statistical evaluation of this correlation, based on our review of critical heat flux correlations, considering both uniform and non-uniform axial heat flux data, we consider that changes are possible. If the results of the non-uniform DNB tests are not available

when the technical specifications for Diablo Canyon are finalized, we will require that the minimum allowable DNBR be increased 5 percent above that required to satisfy the 95/95 criterion.

The Westinghouse Topical Report WCAP-8185 itemized the presently available DNB margin in the reference 17 x 17 fuel design. The sources and amounts of these margins for a four loop plant are as follows:

Source	DNBR Margin(%)
DNB calculations used a multiplier of 0.86 while data justify a multiplier of 0.88.	2
A DNBR of 1.3 was used in lieu of the 95/95 criterion. Data justifies a DNBR of 1.275.	2
A TDC value of 0.051 was used in the data reduction while a value of 0.038 was applied in the analysis.	1.4
DNB tests were performed with 26 in. grid spacing while the design utilized 20.5 in. spacing.	5

Thus, the Diablo Canyon design offers a total DNBR margin of approximately 10 percent beyond the requirements of the Westinghouse criteria. We find this margin to be sufficient to cover uncertainties due to the present state of incomplete results of the 17 x 17 tests and unavailability of as-built tolerances. We therefore have concluded that the thermal and hydraulic performance of the Diablo Canyon design is acceptable for the design conditions shown in Table 4.3.

The reactors for Diablo Canyon Units 1 and 2 were designed to operate at a higher heat output and a higher inlet coolant temperature than Zion Unit 2 (see Table 4.3). This performance increase was partially based on the use of the THINC Code which permitted a more detailed analysis of the thermal and hydraulic characteristics of the core. The THINC code was developed to consider crossflow between adjacent assemblies in the core as well as thermal diffusion between adjacent subchannels in the assembly. The effect of changes in local power distribution on flow redistribution is also considered. As a result of these considerations, the THINC code permits the computation of more realistic power shapes than those that had been available from previously used computer codes. These power shapes are especially important at the design overpower conditions.

Certain elements of the THINC verification test program have been performed at the Zion facility this year. We will review the results of these tests and analyses as they become available. In the event that sufficient verification cannot be obtained from the combined test and analytical programs, restrictions will be imposed on the operation of the Diablo Canyon reactors. Changes to the technical specifications will be made to maintain required margins to fuel rod damage during normal operation, as well as during anticipated transients.

Another parameter that influences the thermal-hydraulic design of the core is rod-to-rod bowing with fuel assemblies. Experimental data

on the extent of bowing in the 17 x 17 design are not yet available. The 17 x 17 fuel performance surveillance program should provide this information. In the meantime, the design of the core is based on predicted values of bowing derived from measurements made on incore 15 x 15 fuel assemblies. Westinghouse recently submitted two topical reports, WCAP-8346, "An Evaluation of Fuel Rod Bowing," and WCAP-8176, "Effect of a Bowed Rod on DNB," that describe the analytical techniques used to predict bowing and the methods used for assessing the effect of bowing on thermal performance. We are presently reviewing these Westinghouse Topical Reports and will report the results of our evaluation in a supplement to this Safety Evaluation Report.

We have concluded, subject to satisfactory resolution of the matters discussed above, that the thermal and hydraulic design of the Diablo Canyon reactors is acceptable, and that these reactors can operate at core power levels of 3328 and 3411 MWt for Units 1 and 2, respectively. Resolution of the outstanding items disussed in this section will be reported on in a supplement to this Safety Evaluation Report prior to a decision concerning the issuance of operating licenses for Diablo Canyon Units 1 and 2.

Table 4.1

Fuel Mechanical Design Comparison

Design Parameter	Westinghouse Diablo Canyon Plant	Westinghouse D. C. Cook Plant
FUEL ASSEMBLY		
Rod Array	17x17	15x15
Number of Fueled Rods	264	204
Number of Spacer Grids	8	7
Number of Guide Thimbles	24	20
Inter-rod Pitch	.496 in	.563 in
Average Thermal Output (4 loop)	5.4 kW/ft	7.0 kW/ft
FUEL PELLETS		
Density (theoretical)	95%	94%
Fuel Weight/Unit Length	.364 lbs/ft	.462 lbs/ft
FUEL CLADDING		
Outside Radius	.187 in	.211 in
Thickness	.0225 in	.0243 in
Radius/Thickness Ratio	8.31	8.68

Table 4.2

Range Of Design Parameter Experience

PARAMETER	RANGE ON POWER REACTOR EXPERIENCE
Fuel Rod Array	14x14, 15x15
Rod Assembly	179 to 204
Guide Thimbles/Assembly	16 to 20
Assembly Envelope	7.76 in to 8.43 in
Inter-rod pitch	.556 in to .563 in
Plenum length	3.27 in to 6.69 in
Prepressurization	14.7 psia to >400 psia
Diametral gap	.0065 in to .0075 in
Spacer Grids/Assembly	7 to 9

Table 4.3

Thermal And Hydraulic Design Parameters

	Diablo Canyon 17 x 17 With Densification Unit #1	Diablo Canyon 17 x 17 With Densification Unit #2	Diablo Canyon 15x15 Without Densification Unit #1	Diablo Canyon 15x15 Without Densification Unit #2	Zion Unit #2 15x15 Without Densification
Reactor Core Heat Output, MWt	3338	3411	3338		3250
System Pressure, Nominal, psia	2250	2250	2250	2250	2250
Minimum DNBR at Nominal Conditions					
Typical Flow Channel	2.31	2.24	2.08	1.98	2.02
Thimble (Cold Wall) Flow Channel	1.87	1.82		Not Available	Not Available
Total Thermal Flow Rate, lb/hr	132.9×10^6	134.0×10^6	134.1×10^6	134.0×10^6	135.0×10^6
Effective Flow Rate for Heat Transfer, lb/hr	126.9×10^6	127.9×10^6	128.1×10^6	128.0×10^6	128.9×10^6
Effective Core Flow Area, Ft^2	51.1	51.1	51.2	51.2	51.4
Average Velocity Along Fuel Rods, ft/sec	15.4	15.6	15.5	15.6	15.3
Coolant Temperature					
Nominal Inlet, °F	544.4	545.0	544.4	545.0	530.2
Average Rise in Core, °F	67.1	67.8	66.6	67.8	66.8

Table 4.3 (Cont'd)

	Diablo Canyon 17x17 With Densification Unit #1	Diablo Canyon 17x17 With Densification Unit #2	Diablo Canyon 15x15 Without Densification Unit #1	Diablo Canyon 15x15 Without Densification Unit #2	Zion Unit #2 15x15 Without Densification
Active Heat Transfer Surface Area, ft^2	59,700	59,700	52,200	52,200	52,200
Average Heat Flux, Btu/ $-ft^2$	185,700	189,300	212,600	217,200	207,900
Maximum Heat Flux for normal operation, $Btu/hr-ft^2$	464,300	474,500	580,000	580,000	579,600
Average Thermal Output, kW/ft	5.33	5.44	6.88	7.03	7.0
Maximum Thermal Output for normal operation, kW/ft	13.3	13.6	16.9	16.9	18.8(1)
Heat Flux Hot Channel Factor, F_Q	2.5	2.5	2.4	2.4	2.5
Peak Fuel Central Temperature at 100% Power, °F	3400	3400	4250	4250	4250

(1) The value of 18.8 kw/ft tabulated for the Zion plant corresponds to the design limit appropriate to the Westinghouse PWR's prior to the advent of the Interim Acceptance Criteria for Emergency Core Cooling. The application of these criteria result in a requirement for a more uniform power distribution. Thus, the peak linear heat generation rate at 100% of rated power is specified as 16.9 kw/ft for the Diablo Canyon Plant.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

The reactor coolant system (RCS) for the Diablo Canyon Units consists of four coolant loops connected to the reactor pressure vessel Each loop contains a reactor coolant pump, steam generator and associated piping and valves. Reactor coolant system pressure will be maintained by a pressurizer connected to one of the coolant loop hot legs. All of these components are located within the containment building.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

We have reviewed the pressure-retaining components within the reactor coolant pressure boundary (RCPB). In accordance with Section 50.55a of 10 CFR Part 50, these components have been identified and classified into three categories: (1) ASME Section III, Code Class A; (2) ANSI B31.7, Class 1; and (3) ANSI B16.5. These components were designed and constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards. The specified ASME Section III code cases, whose requirements have been applied in the construction of pressure-retaining, Code Class A components within the RCPB (Quality Group Classification A), are acceptable.

We have concluded that compliance with the requirements of the code cases mentioned above, in conformance with the Commission's regulations, provides reasonable assurance that the resulting quality standards and

classifications are commensurate with the importance of the safety function of the reactor coolant pressure boundary. Therefore, we conclude that the applicant's program regarding code cases applicable to the RCPB is acceptable.

The design loading combinations specified for RCPB components are comparable to the plant conditions currently identified as normal, upset, emergency or faulted. The design limits used by the applicant for these plant conditions are comparable to the criteria recommended in Regulatory Guide 1.48. Use of these criteria for the design of the RCPB components provides reasonable assurance that, in the event (1) an earthquake should occur at the site, or (2) a system upset, emergency or faulted transient should occur during normal plant operation, the resulting combined stresses imposed on the system components may be expected not to exceed the allowable design stresses and strain limits for the materials of construction. Limiting the stresses and strains under such loading combinations provides a basis for the design of the system components to withstand the most adverse loadings postulated to occur during the service lifetime without gross loss of the system's structural integrity. The load combinations and associated stress and deformation limits considered in the design of RCPB components constitute an acceptable basis for design in satisfying the related requirements of AEC General Design Criteria Nos. 1, 2, and 4.

5.2.2 Overpressurization Protection

Overpressurization protection in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Article 9, 65th Edition, is provided by pressure relief of the RCS using three pressurizer safety valves mounted on the pressurizer nozzles. The pressurizer safety valves discharge to the pressurizer relief tank. Each of the safety valves is rated to carry 420,000 lbs/hr., which is greater than one-third of the total rated capacity of the system. The maximum pressure transient imposed on the RCS results from a power imbalance caused by a turbine trip from maximum overpower conditions.

We find the method of overpressurization protection to be acceptable.

5.2.3 General Material Considerations

We have reviewed the materials of construction for the reactor coolant pressure boundary to assure that the possibility of serious corrosion or stress corrosion is minimized. All materials used are compatible with the expected environment, as proven by extensive testing and satisfactory service performance. The possibility of intergranular stress corrosion in austenitic stainless steel used for components of the RCPB will be minimized because sensitization has been avoided and adequate precautions have been taken to prevent contamination during manufacture, shipping, storage, and construction. The measures taken to avoid sensitization are in general conformance with the recommendations of Regulatory Guide 1.44, and include controls on compositions, heat treatments, welding processes,

and cooling rates. The use of materials with satisfactory service experience, and conformance to the recommendations of Regulatory Guide 1.44 provide reasonable assurance that austentic stainless steel components will be compatible with the expected service environments, and that the probability of loss of structural integrity is minimized.

Further protection against corrosion problems will be provided by control of the chemical environment. The composition of the reactor coolant will be controlled, and the proposed maximum contaminant levels, as well as the proposed pH, hydrogen overpressure, and boric acid concentrations, have been shown by tests and service experience to be adequate to protect against corrosion and stress corrosion problems.

We have reviewed the controls proposed to prevent hot cracking (microfissuring) of austenitic stainless steel welds. These precautions include control of weld metal composition and welding processes to ensure that the filler metals contain from 5 to 15 percent delta ferrite, as calculated from a Schaeffler constitution diagram. The interpass temperature of all welding methods will be limited to a maximum temperature of 350°F. The proposed welding materials, procedures, and methods comply with Sections III (Subsection NB2432) and IX of the ASME Boiler and Pressure Vessel Code. The use of materials, processes, and test methods that are in accordance with these requirements and recommendations will provide reasonable assurance that loss of integrity of austenitic stainless steel welds caused by either hot cracking or sensitization during welding will not occur.

5.2.4 Fracture Toughness

We have reviewed the materials selection, toughness requirements, and the extent of materials testing proposed by the applicant to provide assurance that the ferritic materials used for pressure retaining components of the RCPB will have adequate toughness under test, normal operation, and transient conditions. The ferritic materials will meet the test requirements and acceptance standards of the ASME Code, Section III, Subsection NB2300 of the Summer 1972 Addenda. These materials also meet the requirements of Appendix G of 10 CFR Part 50. The fracture toughness tests and procedures required by Section III of the ASME Code as augmented by Appendix G of 10 CFR Part 50 for the reactor vessel provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure retaining components of the RCPB.

The reactor will be operated in a manner to minimize the possibility of rapidly propagating failure, in accordance with Appendix G to Section III of the ASME Code, Summer 1972 Addenda, and Appendix G of 10 CFR Part 50. Additional conservatism in the pressure-temperature limits used for heatup, cooldown, testing, and core operation will be provided. These will be determined assuming that the beltline region of the reactor vessel has already been irradiated. The use of Appendix G of the ASME Code as a guide for establishing safe operating limitations, and the use of results of fracture toughness tests performed in accordance with the code and AEC regulations, will ensure adequate safety margins during operating, testing,

maintenance, and postulated accident conditions. Compliance with these code provisions and AEC regulations constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion No. 31.

The toughness properties of the reactor vessel beltline material will be monitored throughout the service life with a material surveillance program that will meet the requirements of Appendix H of 10 CFR Part 50 (July 17, 1973). The specimen orientation and number of specimens per capsule conform to ASTM E 185-70 for Unit 1; this standard was in effect when the vessel was manufactured. For Unit 2, the specimen orientation, number, selection procedure, and removal schedule conform to ASTM F 185-73. Changes in the fracture toughness of material in the reactor vessel beltline caused by exposure to neutron irradiation will be assessed properly, and adequate safety margins against the possibility of vessel failure can be provided if the material surveillance requirements of ASTM E 185-70, ASTM E 185-73, and Appendix H to 10 CFR Part 50 are met. Compliance with these documents will ensure that the surveillance program constitutes an acceptable basis for monitoring irradiation induced changes in the fracture toughness of the reactor vessel material, and will satisfy the requirements of AEC General Design Criterion No. 31.

5.2.5 Pump Flywheel

The probability of a loss of pump flywheel integrity has been minimized by the use of suitable material, adequate design, and inservice inspection. The applicant has stated that the integrity of the reactor coolant pump flywheel is assured by compliance with Regulatory Guide 1.14,

"Reactor Coolant Pump Flywheel Integrity." The use of suitable material, and adequate design and inservice inspection for the flywheels of reactor coolant pump motors, as specified in the FSAR, provide reasonable assurance (1) that the structural integrity of flywheels is adequate to withstand the forces imposed in the event of pump design overspeed transient without loss of function, and (2) that their integrity will be verified periodically in service to assure that the required level of soundness of the flywheel material is adequate to preclude failure. Compliance with the recommendations of AEC Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion No. 4.

5.2.6 Pump Overspeed

The Regulatory staff is investigating, on a generic basis, the consequences of an unlikely rupture of a reactor coolant pipe which in certain locations might result in reactor coolant pump overspeed. If this study indicates that additional protective measures are warranted under specific circumstances in order to prevent significant pump overspeed or to limit the potential effects on safety related equipment, the staff will review the circumstances applicable to the Diablo Canyon Units to determine what modifications, if any, are required to assure that an acceptable level of safety is maintained.

5.2.7 RCPB Leakage Detection System

Adequate provisions have been made to detect leakage of reactor coolant to the containment. The major components of the system are: containment atmosphere particulate radioactivity monitors, radiogas monitors, level

indicators on the containment sumps, and a water temperature monitor on the containment air recirculation and cooling unit. The system has sufficient sensitivity to measure small leaks, will identify the leakage source to the extent practicable, will include suitable control alarms and readouts, and conforms with the functional requirements recommended in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." In addition, indirect indications of leakage can be obtained from the containment humidity, press , and temperature indicators. Significant intersystem leakage will be indicated by abnormal readings from the radioactivity monitors used to detect failed fuel, and indirectly, by the coolant flow and level measuring instrumentation provided for normal operational control of the system.

The leakage detection systems have detection capabilities which conform with those recommended in Regulatory Guide 1.45, and provide reasonable assurance that any structural degradation resulting in leakage during service will be detected in time to permit corrective actions. This constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion No. 30.

5.2.8 Inservice Inspection Program

The program for the inservice inspection of the reactor coolant system will be conducted, to the extent practical, in compliance with the ASME Boiler and Pressure Vessel Code, Section XI, 1971 Edition, including the Winter 1972 Addenda. The program will also conform to Regulatory Guide 1.51, "Inservice Inspection of ASME Code Class 2 and

3 Nuclear Power Plant Components," for the inspection of Class 2 systems. Remote access methods and data acquisition methods have been developed to facilitate the inspection of those areas of the reactor vessel not readily accessible to inspection personnel. The conduct of periodic inspections and hydrostatic testing of selected welds and weld heat-affected zones of the pressure retaining components in the RCPB in accordance with the requirements of Section XI of the ASME Code provides reasonable assurance that evidence of structural degradation or loss of leaktight-integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of AEC General Design Criterion No. 32.

5.3 Reactor Vessel Integrity

We have reviewed all factors contributing to the structural integrity of the reactor vessels and conclude that there are no special considerations that make it necessary to consider potential vessel failure for Units 1 and 2 of the Diablo Canyon Plant. The bases for our conclusion are that design, material, fabrication, inspection and quality assurance requirements conform to the rules of the ASME Code, Section III, 1965 Edition, including the Summer 1966 Addenda for Unit 1 and the 1968 Edition for Unit 2.

Although the applicable code cases were not listed in the FSAR, the applicant has stated that an effort was maintained during construction

to continually update and apply AEC Quality Standards to the requirements of the vessels. We find this to be acceptable.

The stringent fracture toughness requirements of the ASME Code, Section III, 1971 Edition, and the 1972 Summer Addenda will be met. Also, operating limitations on temperature and pressure will be established for these plants in accordance with Appendix G ("Protection Against Non-Ductile Failure") of the 1972 Summer Addenda of Section III of the ASME Code.

The integrity of the reactor vessels is assured because the vessels:

1. Were designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and pertinent code cases.
2. Were made from materials of controlled and demonstrated high quality.
3. Were inspected and tested to provide substantial assurance that the vessels will not fail because of material or fabrication deficiencies.
4. Will be operated under conditions and procedures and with protective devices that provide assurance that the reactor vessel design conditions will not be exceeded during normal reactor operation or during most upsets in operation, and that the vessels will not fail under the conditions of any of the postulated accidents.
5. Will be subjected to monitoring and periodic inspection to demonstrate that the high initial quality of the reactor vessels has not deteriorated significantly under the service conditions.
6. May be annealed to restore the material toughness properties if this becomes necessary.

5.4 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts and bolts have become loose parts within reactor coolant systems. In addition to causing operational inconvenience, such loose parts can damage other components within the system or be an indication of undue wear or vibration. For such reasons, the staff has encouraged applicants over the past several years to support programs designed to develop effective, on-line loose parts monitoring. For the past few years we have required each applicant for an operating license of a PWR plant to initiate a program, or to participate in an ongoing program, the objective of which is the development of a functional, loose parts monitoring system within a reasonable period of time. Recently, prototype loose parts monitoring systems have been developed and are presently in operation or being installed at several plants. The applicant has selected an appropriate available monitoring system, and has made a commitment to install this system prior to plant operation. This commitment to provide a loose parts monitor is acceptable to the staff.

5.5 Residual Heat Removal System

The residual heat removal system (RHRS) is designed to remove decay heat and sensible heat from the RCS and core during the latter stages of cooldown. The system also maintains the reactor coolant temperature during refueling, provides the means for filling and draining the refueling cavity, and as a secondary function, serves as part of the ECCS during the injection and recirculation phases of a LOCA. The RHRS consists of two heat

exchangers, two pumps, and associated piping, valves, and instrumentation required for operational control.

The RHRS is placed into operation approximately 4 hours after initation of plant shutdown when the temperature and pressure of the RCS are below 350°F and 425 psig, respectively. Assuming operation of the two pumps and two heat exchangers, and that each heat exchanger is supplied with component cooling water at design flow and temperature, the RHRS is designed to reduce the RCS temperature from 350°F to 140°F within 16 hours. Since the RHRS is provided with two pumps and two heat exchangers arranged in separate independent flow paths, if one of the two pumps or one of the two heat exchangers is not operable, safe cooldown of the plant is assured, but the time required for cooldown is extended. (See Section 7.6 of this report for a discussion of RHRS overpressure interlocks.)

6.2 Containment Systems

6.2.1 Containment Functional Design

The containment systems for each of the Diablo Canyon Units include a reactor containment structure, containment heat removal systems, containment isolation systems, and a combustible gas control system.

The containment is a steel-lined, reinforced concrete structure with net free volume of 2,630,000 cubic feet. The containment structure houses the nuclear steam supply system, including the reactor, steam generators, reactor coolant pumps and pressurizer, as well as certain components of the plant's engineered safety feature systems. The structure is designed for an internal pressure of 47 psig and a temperature of 271°F.

The applicant has described in the FSAR the methods used to analyze the containment pressure response for a spectrum of design basis loss-of-coolant accidents, and the results of these analyses. Various break locations and sizes were evaluated to determine that the double-ended pipe rupture at the pump suction of the reactor coolant system results in the highest containment pressure.

The containment pressure response from postulated loss-of-coolant accidents was analyzed in the following manner. Mass and energy release rates to the containment were calculated and then used as inputs to the COCO computer program to calculate the containment pressure response. The SATAN V computer code was used to determine the mass and energy addition rates to the containment during the blowdown phase of the accident; i.e., the phase of the accident during which most of the

energy contained in the reactor coolant system including the primary coolant, metal, and core stored energy, is released to the containment. To obtain a conservatively high energy release rate to the containment during the blowdown phase, the core was assumed to remain in nucleate boiling for an extended period of time so that the energy release rate from the core would be maximized. Under this assumption, more heat is transferred from the core to the containment than would be predicted by a calculation suitable for core heatup and an emergency core cooling performance evaluation. This additional energy release from the core increases the calculated containment pressure and therefore assures a margin of conservatism in the analysis. The SATAN V computer code has been accepted by the staff for calculating energy released during a LOCA.

During the core reflood phase of the accident, when the core is again filled with water, mass and energy release rates were calculated using a hydraulic model and an energy balance model. The hydraulic model determines the core flooding rate and the entrainment fraction. The energy balance model calculates the core exit conditions and the energy addition from the steam generator. The analysis of the reflood phase of the accident is important with regard to pipe ruptures of the reactor coolant system cold legs, since the steam and entrained liquid carried out of the core for these break locations pass through the steam generators which constitute an additional energy source. The steam and

entrained water leaving the core and passing through the steam generators will be evaporated and/or superheated to the temperature of the steam generator secondary fluid.

Results of the FLECHT experiments indicate that the carryout fraction of fluid leaving the core during reflood is about 80% of the incoming flow to the core, and continues until the fuel is recovered with water to about the 8-foot elevation, at which time the fuel clad temperature transient ceases (quenching occurs). The applicant has conservatively assumed quenching of the core at the 10-foot elevation for the containment pressure calculations.

The rate of energy release to the containment during the reflood phase is proportional to the flow rate into the core. The rupture of the cold leg at the pump suction results in the highest mass flow through the core, and thus through the steam generators. We have compared the mass and energy releases to the containment during the reflood phase of the accident, as calculated with our FLOOD computer code, with those values predicted by the applicant. The results of this comparison indicate equivalent predictions of energy release. Therefore, we have accepted the applicant's computer models as a method of computing core reflood for this plant.

The applicant has included consideration of a possible additional energy release to the containment during the post-reflood phase of the large break accident. This phase begins after the core has been completely covered with water. During this phase, decay heat generation

will produce boiling in the core, and a two-phase mixture of steam and water will exist. In calculations performed by the FROTH code, this two-phase mixture was assumed to rise above the core and enter the steam generators. By this process the remainder of the available steam generator energy was removed by boiling of the water entrained in the two-phase mixture and carried into the containment as steam. In calculating the rate of energy removed from the steam generators, the maximum steam flow based on the hydraulic resistance of the system was used. A portion of the steam that flows through the unbroken loops through the ECCS injection points is assumed to be quenched before exiting to the containment. In the post-reflood period the ECCS system was conservatively assumed to be functioning at one half capacity; this assumption minimizes the condensing effects of quenching and therefore increases the energy released to the containment. We therefore conclude that the applicant has calculated the energy release from the steam generator in a conservative manner.

The COCO code was used to calculate the containment pressure from the mass and energy data discussed above. The peak containment pressure of 43.8 psig was calculated by the applicant assuming maximum operation of the safety injection system and a single-active failure of one spray pump. We have calculated a peak containment pressure of 44.6 psig using the CONTEMPT code. We therefore conclude that the maximum containment pressure is conservatively calculated to be below the design pressure of 47 psig.

The applicant has also analyzed the containment pressure response due to a postulated failure of a main steam line within containment. The maximum calculated containment pressure is about 42 psig which is below the design value for the containment.

The applicant used the TMD (Transient Mass Distribution) code with augmented flow correlation to analyze the transient pressure response of the containment interior compartments. The TMD code was developed by Westinghouse to analyze the short-term pressure response in the containment subcompartments. TMD calculates the critical flow of a two-component, two-phase fluid (air, steam and water), assuming a thermal equilibrium condition. However, a correction factor, which was determined by Westinghouse from data obtained from small scale experiments, is then applied to the calculated critical flow. The correction factor as used in the code increases the critical flow up to 20% through the compartments as the quality of the fluid decreases. Westinghouse refers to this increased critical flow as "augmented" flow. The net effect results in a lower subcompartment differential pressure when compared to the pressure calculated without the augmented flow correlation. The use of the augmented flow factor is therefore less conservative than the use of the thermal equilibrium assumption in calculating subcompartment pressure transients.

During the course of our review of the FSAR, we informed the applicant that we would base our evaluation of the containment subcompartment pressure response on the analysis without the augmented critical flow correlation, and that the subcompartment pressure response should be reanalyzed

with the non-augmented flow correlation. In response to our request the applicant is reanalyzing the subcompartment pressure responses considering a non-augmented flow correlation. The applicant has provided us with an analysis of the pressure response within the pressurizer enclosure and loop compartments using the non-augmented vent flow correlations. We are currently reviewing the applicant's analyses of these subcompartments. The applicant has not completed the analysis of the reactor coolant pipe annulus, reactor vessel annulus, and lower reactor cavity. We expect to receive the results of these analyses at a later time, and will report the results of our evaluation in a supplement to this Safety Evaluation Report.

We have reviewed the design of the components and substructures inside the containment, and subject to favorable resolution of the subcompartment pressure analysis item discussed previously, we find that this design satisfies the requirements of AEC General Design Criteria Nos. 16 and 50, and is acceptable.

6.2.2 Containment Heat Removal Systems

The containment heat removal systems include two redundant containment spray trains and five containment fan-cooling units.

The containment spray system serves only as an engineered safety feature and performs no normal operating function. It is a Seismic Category I system consisting of redundant piping, valves, pumps and spray headers. All active components of the system are located outside the containment. Missile protection is provided by direct shielding or physical separation of equipment. The containment sump intakes

for the spray pumps are covered by a screen assembly designed to prevent debris that could clog the spray nozzles from entering the spray system. A high-high containment building pressure on two of four sensors will cause the ESF actuation system to automatically place the containment sprays in operation. The spray pumps and valves can also be operated manually from the control room. The spray pumps will initially take suction from the refueling water storage tank (RWST). When the water in the RWST reaches a low-low level, the spray pump suction is transferred to the containment sump to initiate the spray recirculation phase. The applicant's analysis indicates that sufficient water will have been delivered to the containment at that time to provide the required NPSH to the spray pumps. The system is designed to provide adequate net positive suction head to the system pumps, considering the water temperatures and containment pressure calculated to be present during the accident. We conclude that the design meets the intent of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps."

The containment fan-cooler system consists of five equal capacity fan-cooler units. Each fan-cooler contains a set of cooling coils, HEPA filters and a two-speed fan. Cooling water for the units is supplied from the component cooling water system. During normal operation, four of the five fan-coolers operating at high speed are required to provide sufficient cooling. Upon receipt of a safety injection actuation signal, the idle fan-cooling unit is automatically started on

the low speed setting. Simultaneously, the running units are switched from high speed to low speed operation. The containment fan-cooling system is a Seismic Category I system. The fan-cooling units are located outside the secondary concrete shield for missile protection, and are accessible for periodic testing and inspection during normal plant operation.

We have reviewed the materials selection for the containment heat removal and ECCS systems, in conjunction with the expected chemistry of the cooling and containment spray system water. The applicant has stated that the use of sensitized stainless steel will be avoided, and that the proposed chemistry will not cause stress corrosion cracking of austenitic stainless steel under conditions that would be present during accident conditions. We have concluded that the controls on material and cooling water chemistry proposed will provide reasonable assurance that the integrity of components of these systems will not be impaired by corrosion or stress corrosion.

We have reviewed the containment heat removal systems for conformance with AEC General Design Criteria Nos. 38, 39, and 40, and have found them to be acceptable.

2.3 Containment Air Purification and Cleanup Systems

The containment air purification and cleanup systems consist of: (1) the normal containment preaccess filtration system; (2) the normal containment purge system; (3) the containment fan-cooler system; and (4) the containment spray additive system.

The preaccess filtration system consists of two fan filter trains, each capable of processing containment atmosphere at 12,000 cfm through a filter bank consisting of a pre-filter, HEPA filter, and a charcoal filter to reduce airborne activity so as to permit safe and continuous access to the containment. This system is not required for post-accident operation.

The normal containment purge system supplies outside air to the containment where it is circulated and then exhausted though pre-filters and HEPA filters. It is designed for use during normal plant operation and serves no post-accident function.

The containment fan-cooler system is provided with HEPA filters to reduce the airborne particulate fission products following a LOCA. Under normal conditions there is no treatment of the air within the containment other than cooling. Upon receipt of a safety injection actuation signal the dampers on the fan-cooler units will be positioned to allow the air-steam mixture to circulate through the HEPA filters, in addition to the cooling coils. The system components are Seismic Category I and are designed to conform to the requirements of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Plants."

In addition to its heat removal function, the containment spray system also is used for iodine removal from the containment atmosphere following a postulated LOCA. Sodium hydroxide is added to the containment spray

solution by the spray additive system to enhance the iodine scrubbing function of the system. The spray additive system consists of the spray additive tank, eductors, valves and connecting piping.

A sufficient quantity of NaOH will be injected to raise the equilibrium pH in the containment sump to a minimum value of 8.5. We have evaluated the containment spray and spray additive systems and found them effective for removal of elemental iodine, and iodine absorbed on airborne particulate matter. The first order removal coefficients for elemental and particulate iodine are 10 and 0.45 (hrs^{-1}), respectively, in an estimated effective volume of 2.16×10^6 ft^3. The minimum sump pH of 8.5 is considered adequate to achieve and maintain a decontamination factor (DF) of 100 for the elemental iodine.

We have reviewed the containment air purification and cleanup systems for conformance with AEC General Design Criteria Nos. 41, 42, 43, and 46, and have found them to be acceptable.

6.2.4 Containment Isolation Systems

The containment isolation systems are designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided so that no single valve or piping failure results in loss of containment integrity. Reactor building penetration piping up to and including the external isolation valve is designed to Seismic Category I requirements, and is protected against missiles that could be generated under accident conditions.

Reactor building isolation will occur automatically upon receipt of a containment isolation signal. All fluid penetrations not required for operation of the engineered safety features equipment will be isolated. Remotely operated isolation valves will have position indication in the control room.

We have reviewed the containment isolation systems for conformance with AEC General Design Criteria Nos. 55, 56 and 57, and have found them to be acceptable.

6.2.5 Combustible Gas Control Systems

Following a LOCA, hydrogen may accumulate inside the containment. The major sources of hydrogen generation include: (1) a chemical reaction between the zirconium fuel rod cladding and water; (2) corrosion of materials of construction; and (3) radiolysis of aqueous solutions in the reactor core and the containment sump. The applicant's analysis of post-LOCA hydrogen generation, which is consistent with the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-Of-Coolant Accident," indicates that the hydrogen concentration in the containment would not reach the lower flammability limit of 4 volume percent (v/o) until about 40 days after the postulated LOCA. We have performed a similar analysis of hydrogen generation in the containment following a LOCA and our results are in agreement with the applicant's.

The containment fan cooler system will provide mixing of the containment atmosphere following a LOCA so as to prevent possible problems

associated with hydrogen stratification. The containment hydrogen purge system, consisting of two redundant purge supply routes, is provided to limit the hydrogen concentrations to below the guideline values given in Regulatory Guide 1.7. The system incorporates several design features that are intended to assure the capability of the system to be operable in the unlikely event of an accident. These features include Seismic Category I design, and redundance to the extent that no single component failure disables the system. Redundant monitoring systems are provided to allow periodic sampling and analysis of the hydrogen concentration in the containment.

Based on our review of the systems provided for combustible gas control following a postulated LOCA, we conclude that these systems will meet the recommendations of Regulatory Guide 1.7, are in conformance with AEC General Design Criteria Nos. 41, 42 and 43, and are, therefore, acceptable.

6.2.6 Containment Leakage Testing Program

The proposed containment design includes provisions and features to permit leakage testing in accordance with the requirements of Appendix J of 10 CFR Part 50. The design of containment penetrations and isolation valves permits individual periodic leakage rate testing at the pressure specified in Appendix J. Included are those penetrations that have resilient seals and expansion bellows, i.e., airlocks, emergency hatches, refueling tube blind flanges, hot process line penetrations, and electrical penetrations.

The proposed reactor containment leakage testing program complies with the requirements of Appendix J of 10 CFR Part 50. Such compliance provides adequate assurance that containment leaktight integrity can be verified periodically throughout service lifetime on a timely basis to maintain such leakages within the limits of the technical specifications.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through leak paths will not be in excess of acceptable limits specified for the site. Compliance with the requirements of Appendix J constitutes an acceptable basis for satisfying the requirements of AEC General Design Criteria Nos. 52, 53, and 54.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 Design Bases

The Diablo Canyon ECCS has been designed to provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the reactor coolant piping resulting in loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. The ECCS is also designed to provide cooling in the event of a main steam line break.

The design bases are to prevent fuel cladding damage that would interfere with adequate emergency core cooling and to mitigate the amount of clad-water reaction for any size break up to and including a double ended rupture of the largest primary coolant line. These requirements

are intended to be met even with minimum engineered safeguards available, such as the loss of one emergency power source together with the unavailability of offsite power.

The ECCS subsystems provided are stated to be of such number, diversity, reliability and redundancy that no single failure of ECCS equipment, occurring during a LOCA, will result in inadequate cooling of the reactor core. Each of the ECCS subsystems are designed to function over a specific range of reactor coolant piping system break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe (9.14 ft^2 is the double-ended area).

The proposed design of the ECCS for the Diablo Canyon Plant is the same as that previously reviewed and approved for the Zion Plant. However, on the basis of our evaluation of the application of the single failure criterion to the functional design of the ECCS, we have identified certain locations where a single incorrectly positioned valve could result in total loss of the intended safety function (See Section 7.3.4 of this report for additional discussion of this item). A tabulation of the valves in question can be found in Item 7.12 of a request for additional information sent to the applicant (See Items 43 and 45 in Appendix A of this report). Resolution of this issue can be accomplished by either: (1) including in the technical specifications provisions to remove power from the electrical system to lock certain motor-operated valves in their preferred safety positions; or (2) providing the necessary design modifications to preclude the loss of capability to perform a specified safety function. We have not yet reached agreement with the applicant on this, and will report on the

resolution of the problem in a supplement to this Safety Evaluation Report.

6.3.2 System Design

The ECCS proposed for the Diablo Canyon Plant consists of accumulator tanks and high pressure injection and low pressure injection systems, with provisions for recirculation of the borated coolant after the end of the injection phase. Various combinations of these systems assure core cooling for the complete range of postulated break sizes.

Following a postulated LOCA, the ECCS will operate initially in the passive accumulator injection mode and the active high pressure injection mode, then in the active low pressure injection mode, and subsequently in the recirculation mode. Each of the four accumulator tanks has a total volume of 1350 cubic feet with minimum water volume of 850 cubic feet and 500 cubic feet of nitrogen gas at a normal operating pressure of 650 psig. Each tank is connected to one of the cold legs of the reactor coolant system by a line with two check valves and a normally open motor operated isolation valve in series. A more detailed discussion of these valves is presented in Section 7.3.3 of this report. The high pressure injection mode of operation, upon actuation of a safety injection signal, consists of the operation of two centrifugal charging pumps (rated at 150 gpm each at a design head of 5800 ft) which provide high pressure injection of boric acid solution (via the boron injection tank maintained at 21,000 ppm boron concentration) into the reactor coolant system. Also part of the high pressure injection mode are two safety injection pumps (rated at 425 gpm each at a design head of 2500 ft), which take their suction initially from the refueling water storage tank (350,000 gallons) with a boron concentration of 2000 ppm.

Low pressure injection consists of two residual heat removal pumps (rated at 3000 gpm each at a design head of 375 ft) which take their suction from the refueling water storage tank. For both the high and low head pumps, the system is designed to provide adequate NPSH, considering the water temperatures and containment pressure calculated to be present during the accident.

When a predetermined amount of water in the refueling water storage tank has been injected, suction will be transferred manually to the containment sump for the recirculation mode of operation. Then the ECCS will provide the long-term core cooling requirements by recirculating the spilled reactor coolant collected in the containment sump back to the reactor vessel via the reactor coolant cold legs after it has been cooled in the RHR heat exchangers.

The materials selection for the ECCS is discussed in Section 6.2.2 of this report.

6.3.3 Performance Evaluation

The emergency core cooling system has been designed to deliver fluid to the reactor coolant system in order to control the calculated cladding temperature transient following a postulated pipe break, and for removing decay heat in the recirculation mode.

On January 4, 1974, the Atomic Energy Commission published its decision in the rulemaking proceeding (Docket No. RM-50-1) concerning acceptance criteria for emergency core cooling systems for light-water cooled nuclear power reactors. This decision included the new amendment to 10 CFR Part 50 which incorporates the ruling. The new ruling specifies

that boiling and pressurized light-water nuclear power reactors fueled with uranium oxide pellets within cylindrical Zircaloy cladding that are licensed after December 28, 1974 shall be provided with an emergency core cooling system (ECCS) which shall be designed such that its calculated cooling performance following a postulated loss-of-coolant accident conforms to the criteria set forth in subparagraph (b) of Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR Part 50. The new criteria include the following limits:

(1) The calculated maximum fuel element cladding temperature does not exceed 2200°F.

(2) The calculated total oxidation of the cladding does not exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In addition, paragraph 50.46 states:

ECCS cooling performance shall be calculated in accordance with an

acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covere Appendix K of 10 CFR Part 50, ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models.

This decision on the Final Acceptance Criteria is applicable to Diablo Canyon Units 1 and 2. Accordingly, the applicant has analyzed the performance of the emergency core cooling system in accordance with the criteria set forth in Section 50.46 and Appendix K to 10 CFR Part 50. This analysis was submitted in Amendment 15 to the FSAR dated August 2, 1974. The adequacy of the emergency core cooling system is being evaluated by the staff in light of the Final Acceptance Criteria. Our evaluation of the performance of the emergency core cooling system, and of the applicant's evaluation model, will be reported in a supplement to this Safety Evaluation Report.

6.3.4 Tests and Inspections

The applicant will demonstrate the operability of the ECCS by subjecting all components to preoperational tests, periodic testing, and inservice testing and inspections. The preoperational tests performed fall into three categories:

(1) System actuation tests to verify: (a) the operability of all ECCS valves initiated by the safety injection signal, the phase A containment isolation signal, and the phase B containment isolation signal; (b) the operability of all safeguard pump circuitry

down through the pump breaker control circuits; and (c) the proper operation of all valve interlocks.

(2) Accumulator injection tests to check the accumulator injection lines to verify that the lines are free of obstructions and that the accumulator check valves operate correctly. The applicant will perform a low pressure blowdown of each accumulator to confirm that the line is clear, and to check the operation of the check valves.

(3) Operational testing of all the major pumps. These pumps consist of the charging pumps, the residual heat removal pumps, the containment recirculation pumps, and the safety injection pumps. The applicant will use the results of these tests to evaluate the hydraulic and mechanical performance of these pumps delivering through the flow paths for emergency core cooling. These pumps will operate under both miniflow (through test lines) and full flow (through the actual piping) conditions. By measuring the flow in each pipe, the applicant will make the adjustments necessary to assure that no one branch has an unacceptably low or high resistance. System checks will be made to ascertain that total line resistances are sufficient to prevent excessive runout of the pump. The applicant will be required to show that the minimum acceptable flows as determined for the FSAR analysis are met by the measured total pump flow and relative flow between the branch lines.

The applicant must demonstrate that all ECCS components meet or exceed FSAR analyses values. In addition, in response to a request from the staff, the applicant has evaluated his proposed compliance with the positions stated in Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." With the exception that the auxiliary feedwater pumps will not inject water into the steam generators, the applicant will comply with Regulatory Guide 1.79. We find this to be acceptable.

The applicant will perform routine periodic testing of the ECCS components and all necessary support systems with the plant at power. Valves that are required to operate after a LOCA will be operated through a complete cycle; pumps will be operated individually in this testing on their miniflow lines except the charging pumps, which will be tested by their normal charging function.

Test circuits will be used to periodically check for leakage of reactor coolant back through the accumulator discharge line check valves to ascertain that these valves seat whenever the reactor coolant system pressure is above a preset value. This periodic system testing will also include a visual inspection of pump seals, valve packings, flanged connections, and relief valves to detect leakage.

The inservice inspection program for the ECCS fluid carrying components is included as part of the ASME Code Class 2 and 3 inspection program. This program is described in Section 5.2.8

of this report.

6.3.5 Conclusion

As stated in Sections 6.3.1 and 6.3.2 of this report, the acceptability of the emergency core cooling system is still being evaluated. Specifically (1) the applicant must agree to either lock-out power to certain motor-operated valves or modify the design to render locking out of power unnecessary; and (2) the applicant's ECCS evaluation model and analysis results have not been found to be acceptable. We will report our conclusions regarding acceptability of the ECCS in a supplement to this Safety Evaluation Report.

6.4 Habitability Systems

The applicant proposes to meet the intent of AEC General Design Criterion No. 19 by use of adequate concrete shielding and by installing a 2100 cfm recirculation charcoal filter (having redundant active components in the control room ventilation system. In addition, the construction details of the control room are such as to result in low air infiltration into the control room under isolated conditions. The charcoal filter will be automatically activated upon an accident signal, high radiation signal, or high chlorine signal. We have calculated the potential radiation doses to control room personnel following a LOCA; the resultant doses are within the guidelines of GDC No. 19.

The applicant has indicated that chlorine will be stored onsite for water treatment purposes. Because the effects of an accidental release of chlorine could have an adverse effect on control room habitability, the applicant will provide appropriate chlorine protection devices such as

quick acting chlorine detectors and self contained breathing apparatus. We conclude that these protection devices are acceptable.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The instrumentation and control systems for the Diablo Canyon Nuclear Power Plant, Units 1 & 2, have been evaluated utilizing: (1) the Commission's General Design Criteria (GDC) as published in July 1971; (2) Institute of Electrical and Electronics Engineers (IEEE) Standards, including IEEE Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE Std 279-1971); and (3) applicable Regulatory Guides for power reactors. These criteria have been used as the bases for evaluating the adequacy of these systems.

The designs of the instrument and control systems for the Diablo Canyon Plant are functionally the same as those for Commonwealth Edison's Zion Station. The evaluation of the Diablo Canyon designs concentrated on the equipment qualification, system implementation, applicability of previous generic evaluations, and concerns unique to the Diablo Canyon Plant itself.

A major design change from that presented during the CP review was implemented in the actuation systems for the reactor trip and engineered safety features. The review of these systems is presented in Sections 7.2 and 7.3 of this report.

The results of our review of selected logic and schematic diagrams, and the findings of our initial site visit are reflected in this Safety Evaluation Report. However, additional selected drawings, technical specifications,

BLANK PAGE

and several unresolved items still must be reviewed; these will be reported on in a supplement to this Safety Evaluation Report.

7.2 Reactor Trip System

7.2.1 General

Our review covered the aspects of the protection system which initiates, monitors, bypasses, tests and controls the reactor trip system, including field implementation, logic diagrams and detailed schematics.

7.2.2 Reactor Trip System Actuation Logic

Functionally, the reactor trip system for Diablo Canyon is essentially the same as that evaluated during the CP review. However, the electro-mechanical relay logic has been replaced by solid state logic. The solid state protection system generates signals to open the reactor trip breakers when the required combination of input signals from the analog process system occurs. The design of the solid state protection system was evaluated as a Westinghouse generic item during the review of the Donald C. Cook Nuclear Plant, Units 1 & 2. However, acceptance of the design for other plants is conditioned on verification of the following: (1) seismic and environmental qualification; (2) that the protective functions provided meet the safety criteria for the particular plant; and (3) implementation of the system requirements for preserving the independence of redundant portions of the protection system. We conducted a detailed review of these areas, and supplemented our review with a site visit to examine the physical installation of the electrical equipment.

7.2.2.1 Physical Separation

During the site visit, we found that the physical separation in the solid state protection system racks was inadequate. The input and output wire bundles terminate at a common connector of the isolation board. The center pins of the connector are not used to maintain physical separation; however, there are no physical barriers or protection to separate the wire bundles as they are routed from the connectors. The applicant has agreed to provide physical barriers to separate the input and output wire bundles. Final resolution of this item will be reported on in a supplement to this Safety Evaluation Report.

7.2.2.2 Electrical Isolation

The photodiode isolators used to electrically isolate the safety signals from the non-safety functions, as implemented in the solid state protection system, have not been qualified as acceptable isolation devices. We informed the applicant that we require tests be performed to verify that the photodiode isolators will meet the design basis requirements for system isolation. The applicant has indicated that he will provide the test procedures and results to substantiate the capability and reliability of the photodiodes as isolation devices. Final resolution of this item will be discussed in a supplement to this Safety Evaluation Report.

7.2.2.3 Seismic Qualification

We have concluded from our review that the results of the seismic qualification tests of the solid state protection system are unacceptable.

During the seismic testing of the system, output relays had momentary contact closure, and in some instances, had a permanent change of contact status due to mechanical latching. We have informed the applicant that the staff's position requires that all safety related electrical equipment be designed to withstand the effects of the safe shutdown earthquake without malfunction before, during, or subsequent to a seismic event. We require that the applicant meet the requirements for seismic qualification of the solid state protection system and other safety related electrical equipment (see Section 7.8 of this report).

7.2.2.4 Conclusions

We have concluded that the solid state protection system will be acceptable, providing that: (1) the test procedure and results for qualifying the photodiode isolation devices are acceptable; (2) adequate separation or barriers are provided to separate the input and output wiring; and (3) the seismic qualification program for functional electrical operability of safety related electrical equipment conforms to our requirements (Section 7.8 of this report).

7.2.3 Process Analog System

Functionally, the process analog system for Diablo Canyon is essentially the same as that evaluated during the CP review. This system has been reviewed for the Zion Station, Units 1 & 2, and for Trojan. The staff has concluded that the functional design and implementation of the process analog system is acceptable for

these plants. However, as a result of our initial site visit, we have determined that the physical separation between the protection and control circuits within the protection system process analog racks is unacceptable. The protection system input wiring to the modular isolation amplifier and the output wiring to the control racks are routed in the same wireways. The electrically isolated outputs are also in physical contact with other protection system internal wiring of that channel in the process racks. These outputs terminate in the control racks. The redundant outputs of protection channels, in some instances, terminate in a common control rack. A failure in the control portion of the system could negate protective actions due to lack of physical separation between the inputs and outputs of the isolation amplifiers, and result in loss of protective function.
The CP Safety Evaluation Report for Diablo Canyon Unit 1 (issued in January 1968) noted that the physical and electrical isolation of protection and control instrumentation was not adequate. The CP Safety Evaluation Report for Unit 2 (issued in November 1969) references the same item. The applicant stated, during the CP safety review of Unit 1, that the protection system would be designed to the proposed IEEE Standards at that time (IEEE Std 279-1968). The implementation of the design presented in the FSAR does not meet the requirements of Sections 4.2, 4.6 and 4.7 of IEEE Std 279-1968. The applicant further states that the protection system in the FSAR for Units 1 and 2 meets all requirements of IEEE Std 279-1971; however,

the staff's observations during the site visit indicate that the implementation of this present design does not meet these requirements.

We require that the applicant either: (1) modify the present system to provide a minimum physical separation of six inches, or provide barriers between the control outputs of the isolation amplifiers and the protection system circuitry, including the inputs to the isolation amplifiers; or (2) qualify the present system, as implemented, by testing. The test scope will include all tests performed to qualify the individual isolation amplifiers, and also monitoring of the other protection system input and output channels for signal interference or degradation. We will report the resolution of this item in a supplement to this Safety Evaluation Report.

7.2.4 Testability of Protection Systems

During our review, we considered the adequacy of the proposed testing of the response time of the protection systems. The applicant considered that a sufficient testing program was provided by preoperational tests, together with the response time tests conducted whenever a component affecting response time was replaced. The staff concluded that, until experience with the Diablo Canyon design or other identical designs demonstrates that the protection system response times, including sensor response times, do not change over long intervals of operating experience, the response time testing should be repeated periodically. The applicant subsequently agreed to repeat the system response time tests during each refueling outage, but no less frequently than every

18 months. We have concluded that this is an acceptable testing program. The technical specifications will include a requirement specifying this test program.

7.2.5 Anticipated Transients Without Scram (ATWS)

The Regulatory staff's requirements with respect to ATWS are provided in the staff's technical report, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," WASH-1270, dated September 1973. As applied to the Diablo Canyon Nuclear Plant, Units 1 & 2, these requirements state that changes should be provided to make ATWS consequences acceptable. Unit 1 has been classified by the staff as a "I.C." facility; for this Unit, the applicant will submit an analysis describing and evaluating the consequences of a postulated ATWS. Unit 2 has been classified as a "I.B." facility; for this Unit the applicant has been requested to implement a program to incorporate any design changes necessary to assure that the consequences of anticipated transients would be acceptable in the event of a postulated failure to scram in accordance with Section II.B of Appendix A of WASH-1270. The applicant documented the information required by WASH-1270, for Units 1 and 2, on October 3, 1974. The staff evaluation of this information will be contained in a supplement to this Safety Evaluation Report.

7.2.6 Conclusion

Subject to favorable resolution of the items indicated in Sections 7.2.2.1, 7.2.2.2, 7.2.2.3. 7.2.3, and 7.2.5, we have concluded that

the design of the reactor trip system meets the Commission's requirements and will be acceptable.

7.3 Engineered Safety Feature Actuation Systems

7.3.1 General

Our review covered those aspects of the protection system which initiates, monitors, bypasses, tests, and controls the engineered safety feature (ESF) systems and their auxiliary supporting systems, including field implementation, logic diagrams and detailed schematic drawings. The following sections identify those areas of the design that were changed by the applicant as a result of our review.

7.3.2 ESF Actuation Logic

The ESF actuation logic is part of the solid state protection system discussed in Section 7.2 of this report. The staff's evaluation of the ESF actuation system designs is presented in the Safety Evaluation Report for the Donald C. Cook Nuclear Plant. Subject to the resolution of the items discussed in Section 7.2 of this report, we conclude that the results of our previous evaluations are applicable to the Diablo Canyon Units, and that the ESF actuation logic system is acceptable.

7.3.3 Accumulator Isolation Valves

The design proposed by the applicant for the control circuits for the motor-operated accumulator isolation valves included provisions for the safety injection signal to automatically open the isolation valves.

We concluded that the proposed design of the control circuits did not provide adequate assurance that the accumulator isolation valves would be open when required. The staff's position, which has been applied to recently-licensed plants, is that in addition to the safety-injection signal, the design should include provisions to automatically open the isolation valves when reactor coolant pressure exceeds a preselected value. The applicant has modified the control circuit design to conform with our position regarding automatic opening of the valves. We conclude that this modified design is acceptable.

7.3.4 Changeover from Injection to Recirculation Mode

The designs of the ECCS and containment spray systems require operator action to change over from the injection mode of operation to the recirculation mode before the refueling water storage tank (RWST) is completely emptied following a loss-of-coolant accident. The applicant's original design provided two level instruments to be used to provide indication of the tank level in the control room. Depending on the failure mode of the level instrument, this could result in the operator either tripping both pumps prematurely or failing to trip the pumps.

As a result of our review, the applicant will revise his design to provide automatic tripping of the low pressure injection (RHR) pumps when the RWST level decreases to a specified low level (see Section 5.5 of this report for a description of the RHR system). The automatic tripping of the low pressure injection pumps significantly increases the time available for the reactor operator to complete the changeover from the injection to the recirculation mode of operation, without degrading

the required performance of the core cooling systems below an acceptable level.

The revised design will also include three level instruments. We have informed the applicant that the instrumentation used to provide the signals for automatic tripping of the low pressure injection pumps, and to provide the information that the operator needs to complete the changeover, must meet the requirements of IEEE Std 279-1971.

We have not reached agreement with the applicant concerning the locking-out of power to the motors of valves, where a single incorrectly positioned valve could result in the total loss of the safety function. The difficulty is that some of these valves are presently included in the procedure for changeover to the recirculation mode, and that the applicant believes that the probability of an incorrectly-positioned valve is sufficiently low so as to be considered incredible. (See Section 6.3.1 of this report for additional discussion of this item). We will report the resolution of this item in a supplement to this Safety Evaluation Report.

We have concluded that, subject to the satisfactory resolution of the above mentioned concerns, the operator will have sufficient time to perform the actions required to change over to the recirculation mode of operation, and the design will meet the Commission's requirements and is, therefore, acceptable.

7.4 Systems Required for Safe Shutdown

We have reviewed the instrumentation and control systems being provided for safe shutdown, including the design provisions for achieving

hot and cold shutdown in the event that access to the main control room is restricted or lost. We have concluded that these systems conform to the Commission's requirements and are, therefore, acceptable.

7.5 Safety Related Display Information

The safety related display information provides data to enable the operator to perform the required manual safety functions, and provides information for post-accident surveillance. The instrumentation provided is similar to that for the Zion Nuclear Plant, except for the physical configuration. The applicant has identified the parameters to be monitored during and after an accident or abnormal occurrence. The instrumentation is redundant, qualified for the environment, powered by redundant vital a-c sources, and a minimum of one channel is recorded. We have reviewed the drawings and verified the implementation during the site visit. However, the applicant has not provided a description of the bypass and inoperable status indication in the FSAR. The applicant has indicated that he will provide this description. Resolution of this item will be discussed in a supplement to this Safety Evaluation Report.

We have concluded, subject to documentation by the applicant of the description of the bypass and inoperable status indication, that the safety related display provides the required information indicated above, and that it conforms to the Commission's requirements and is, therefore, acceptable.

7.6 RHR Overpressure Protection Interlocks

The initial design of the interlocks for the motor operated isolation valves on the suction side of the RHR pumps, which are provided to prevent overpressurization of the RHR system by the reactor coolant system, did not meet our criteria. We require interlocks of diverse principles in order to prevent opening of these valves when the reactor coolant system pressure is greater than approximately 425 psig, and to automatically close the valves when the system pressure exceeds approximately 600 psig. The applicant has modified the design to conform with these criteria, and, therefore, we conclude that the design is equivalent to those provided on other recently licensed plants, and is acceptable.

7.7 Control Systems Not Required for Safety

The design of these control systems provided for the Diablo Canyon Plant is similar to those for other recently licensed plants, including the Zion Nuclear Plant, except for the following differences: Zion has analog rod position indication and 50% load rejection capability while Diablo Canyon has digital rod position indication and 100% net load rejection. We have determined from our review that these design differences do not affect the safety of the plant. We conclude that the Diablo Canyon design is acceptable.

7.8 Environmental and Seismic Qualification

We have not completed our review of the environmental qualification of safety related electrical equipment for Diablo Canyon. A major portion of the equipment was qualified by test programs documented in Topical

Report WCAP-7744, "Environmental Testing of Engineered Safety Features Related Equipment (NSSS Standard Scope)." This topical report was reviewed and found to be unacceptable by the Regulatory staff. We therefore conclude that the environmental qualification testing of the NSSS supplied safety related electrical equipment for the Diablo Canyon Plant is unacceptable. The applicant has been informed that we require a response to the concerns indicated in the staff's review of WCAP-7744. The resolution of these concerns will be provided in a supplement to this Safety Evaluation Report.

We have not completed our review of the seismic qualification of safety related electrical equipment that must operate before, during, and subsequent to a seismic event. A major portion of the equipment was qualified by test programs documented in WCAP-8021. Based on the results of these tests, we have concluded that the electrical functional capability of the equipment tested, in that it did not operate as was designed, is unacceptable. The applicant has been informed that we require either a test program and results indicating that all safety related electrical equipment will operate as designed, without malfunction, before, during and subsequent to a seismic event, or else the results of a failure mode and effects analysis of all possible combinations of abnormal equipment operations which demonstrates that the safety of the plant is not compromised during a seismic event. The resolution of this item will be provided in a supplement to this Safety Evaluation Report.

7.9 Conclusion

Subject to the resolution of the items indicated in Sections 7.2.2.1, 7.2.2.2, 7.2.3, 7.2.5, 7.3.2, 7.3.4, 7.5 and 7.8, we have concluded that the design of the electrical instrumentation and controls meet the Commission's requirements, and is, therefore, acceptable.

8.0 ELECTRIC POWER

8.1 General

The Commission's General Design Criteria (GDC) 17 and 18, IEEE Standards including IEEE Criteria for Class IE Electric Systems for Nuclear Power Generating Stations (IEEE Std 308-1971), and Regulatory Guides 1.6, 1.9, 1.32 and 1.41, served as the bases for evaluating the adequacy of the electric power systems of the Diablo Canyon Nuclear Plant, Units 1 & 2.

8.2 Offsite Power

The Diablo Canyon Plant will be interconnected to PG&E's electric grid system via two 230 kV and three 500 kV lines emanating from their respective switchyards; these yards are physically and electrically separated and independent of each other. The two incoming 230 kV transmission lines, which share common towers, provide power to the 230 kV switchyard which is a double bus arrangement with the capability of feeding the two start-up/standby transformers (one per Unit) from either source or bus, isolating the other source. The 230 kV power system provides an immediate source of offsite power. The three incoming [illegible] kV transmission lines, located on two rights-of-way, provide power to the 500 kV switchyard which is a breaker-and-a-half configuration with the capability of backfeeding through the main transformer of either unit. This source of offsite power is made available by manually initiating a motor operated disconnecting link which is an integral part

of the generator's isolated phase bus. This disconnect link is operated from the control room and is interlocked to prevent opening under load. The applicant states that this source of offsite power can be made available by manual initiation in approximately 30 seconds, providing a delayed source of offsite power. The 230 kV and 500 kV switchyards have primary and backup relaying control and independent d-c control power sources for their respective switchyard breakers.

A single line from the 230 kV switchyard supplies the standby/start-up transformer (230 kV/12 kV) of each unit. The low voltage side of these transformers supplies the 12 kV buses and the primary side of another standby/start-up transformer (12 kV/4.16 kV) for each unit. The low voltage side of these transformers supplies the 4.16 kV and the three ESF buses of each unit.

The 500 kV switchyard can supply power by backfeeding through the main transformers (500 kV/25 kV) of each unit to the unit auxiliary transformers. One auxiliary transformer (25 kV/12 kV) per unit supplies the 12 kV buses and a second auxiliary transformer (25 kV/4.16 kV) per unit supplies the 4.16 kV buses and the three ESF buses of each unit. Either the standby/start-up or unit auxiliary transformers and their attendant distribution systems have sufficient capacity to supply shutdown and emergency load requirements.

The applicant has performed an electrical grid stability analysis which indicates that the loss of any single generator on the grid,

including that for Diablo Canyon Units 1 or 2, while operating at full load, will not adversely affect the stability of the remainder of the transmission grid or the ability to provide offsite power to the Diablo Canyon Plant. Furthermore, Diablo Canyon Units 1 and 2 have the capacity of 100% net load rejection without reactor trip.

The design of the offsite electrical power system has provisions for periodic inspection and testing to demonstrate that all Class IE electrical systems are capable of performing their safety functions.

We have concluded that a combination of either 230 kV circuit and one of the three 500 kV circuits provides sufficient assurance that redundant and independent sources of offsite power are provided, as required by AEC General Design Criterion No. 17, and that the design of the offsite power system meets AEC General Design Criteria Nos. 17 and 18, IEEE Std 308-1971, and Regulatory Guide 1.32, and is, therefore, acceptable.

8.3 Onsite Power

8.3.1 A-C Power Systems

The engineered safety feature loads are divided among three 4.16 kV ESF buses per unit. The ESF loads are assigned to these buses such that operability of any two 4.16 kV ESF buses and their attendant distribution systems will supply the minimum safety requirements for that unit. Two sources of offsite power are provided to the three ESF buses per unit, as discussed in Section 8.2 of this report.

The onsite power source for Units 1 and 2 is supplied by five 4.16 kV, 2600 kW diesel generators. Four diesel generators are separately assigned to four ESF buses, two per unit. The fifth diesel generator is assigned

to be shared between the third ESF bus from each unit. The shared diesel generator swings to the unit having a safety injection signal. Undervoltage of an ESF bus will start its respective diesel generator and a safety injection signal will start all of the diesel generators.

The diesel generators for Diablo Canyon have been previously qualified for use in nuclear power plant applications and are being utilized as onsite power sources at several nuclear plants. The onsite system has the capability to provide power for redundant ESF equipment for one unit in an accident condition, and safely shutdown the other unit. The system also has the capability to provide the minimum ESF equipment power requirements for one unit in an accident condition, and safely shut down the other unit, assuming a false accident signal or a single failure. No communication is required between the operators of both units to maintain each unit in a safe condition.

The diesel generators are individually located in separate Seismic Category I compartments on the ground floor of the turbine building. These compartments are protected from fire, flooding, and external and internal missiles. The fuel oil system has two underground fuel oil storage tanks which are shared by both units. The fuel oil system is designed to remain operable after sustaining a single failure of either an active or passive component. The total onsite fuel oil storage capacity provides a minimum of seven days of fuel for diesel generator operation supplying the power requirements of: (1) the minimum ESF equipment for one unit, and (2) the equipment required for maintaining the second unit in a shutdown condition.

The 120 volt a-c vital instrumentation bus system supplies power for instrumentation, control, protection and annunciation. The vital a-c system consists of separate and electrically independent buses and attendant distribution systems, with each supplying their respective redundant load groups. Each bus is served by a separate inverter which is supplied from a 125 volt d-c system or from one of the 480 volt a-c ESF buses; this satisfies the single failure criterion. An additional 120 volt a-c source, fed from a 480 volt ESF bus, is available, through a manual transfer switch, to provide power to the vital instrument buses when an inverter is out of service.

8.3.2 D-C Power Systems

The onsite d-c emergency power systems for each unit consist of 125 and 250 volt systems. Three 125 volt batteries are provided for each unit (batteries 11, 12, and 13 for Unit 1, and batteries 21, 22, and 23 for Unit 2). Each unit has one of these 125 volt batteries assigned to a separate and independent bus (batteries 13 and 23 for Units 1 and 2, respectively). Battery 11 is in series with battery 22 providing a 250 volt d-c system for Unit 1, and battery 12 is in series with battery 21 providing a 250 volt d-c system for Unit 2. The 125 volt d-c sources are shared as described above, between Units 1 and 2; however, there is no interconnection between redundant systems within each unit.

The 125 volt d-c systems provide power for control, instrumentation, annunciation and emergency lighting. The 250 volt d-c systems provide power for the turbine lubricating systems. Each of the 125 volt d-c

buses has a battery charger assigned to it. The two buses shared with the other unit have a backup swing charger that can be manually connected to either bus. The third 125 volt d-c bus has a redundant charger. Each of the five chargers per unit are capable of maintaining the required d-c voltage on the system and recharging the batteries. The battery chargers are supplied by the onsite standby power system in the event of loss of station and offsite power. Separate battery rooms and switchgear rooms are provided for each of the 125 volt d-c distribution systems.

8.3.3 Conclusions

We have reviewed the design of the onsite a-c and d-c power distribution systems, and have determined that the design meets AEC General Design Criteria Nos. 17 and 18, IEEE Std 308-1971, and Regulatory Guides 1.6, 1.9, and 1.32. The applicant will conduct preoperational tests to confirm the independence of redundant onsite power distribution systems, their controls, and loads. These preoperational tests will meet the recommendations of Regulatory Guide 1.41. We conclude that the design of the electrical power systems meets the Commission's requirements, and is, therefore, acceptable.

8.4 Physical Independence of Electrical Equipment and Circuits

Section 8.3.3 of the FSAR provides a description and analysis of the applicant's criteria and procedures for providing physical independence of safety related circuits and equipment. We have reviewed the applicant's criteria and procedures regarding separation, and verified their implementation during our initial site visit. The applicant has not

responded to our request to provide a description and analysis of his criteria for protection of Class 1E cabling and equipment in hazardous and missile prone areas. We will report the resolution of this item in a supplement to this Safety Evaluation Report. With a satisfactory response to this item and resolution of the staff's concerns regarding separation in the process analog system (see Section 7.2.3 of this report), we conclude that the separation criteria meet the Commission's requirements, and are, therefore, acceptable.

9.0 AUXILIARY SYSTEMS

9.1 General

In the course of our review of the auxiliary systems, we have directed our attention to the design of the systems with respect to the safety related objectives of these systems, and to the manner in which these objectives are achieved.

The auxiliary systems necessary to assure reactor safety are: (1) auxiliary saltwater system, (2) component cooling water system, (3) portions of the makeup water system, (4) condensate storage facilities, (5) ultimate heat sink (in conjunction with the auxiliary saltwater system and the intake structure), (6) control room heating, ventilation, and air conditioning system, (7) auxiliary building heating, ventilation, and air conditioning system, (8) fuel handling area heating and ventilation system, (9) fire protection system, and (10) diesel generator auxiliary systems. The systems necessary to assure safe handling and adequate cooling of the spent fuel include the fuel handling systems and the spent fuel pool cooling and cleaning system.

We have also reviewed those auxiliary systems whose failure would not prevent safe reactor shutdown, but could, either directly or indirectly, be the potential source of a radiological release to the environment. These systems include the floor drainage and sampling system. From our review of the proposed design of the floor drainage and sampling systems

BLANK PAGE

for the Diablo Canyon Units, we find that they are comparable in design and function to other PWR facilities that have been previously reviewed and approved. On this basis, we have concluded that these auxiliary systems are in compliance with the applicable rules and regulations and are acceptable.

We have reviewed those systems and components to be shared by both Units, and find that their designs meet the requirements of AEC General Design Criterion No. 5, and are acceptable.

9.2 Fuel Storage and Handling

9.2.1 New and Spent Fuel Storage

Fuel handling and storage facilities for each unit are provided for storage and transfer of new and spent fuel. Spent fuel will be stored underwater in the spent fuel storage pool and new fuel will be stored dry in the new fuel storage area. The new and spent fuel storage racks are designed to prevent assemblies from being placed in other than prescribed locations. The new fuel storage area accomodates one-third of a core and the spent fuel pool accomodates a full core plus the normal quantity of spent fuel assemblies from the reactor during refueling (usually one-third of a core). The fuel is stored in a vertical array with sufficient center-to-center distance between assemblies to assure that k_{eff} will not exceed 0.90. even if unborated water is used to fill the pool.

We conclude that the designs of the new and spent fuel storage facilities are acceptable.

9.2.2 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling system is designed to remove heat generated by stored spent fuel elements in the spent fuel pool. A secondary function of this system is to clarify and purify spent fuel pool water, transfer pool water, and refueling water. The system is designed and constructed to Seismic Category II requirements; however, demineralized makeup water can be added to the pool from Seismic Category I sources. Each Unit has an independent cooling and cleanup system for its spent fuel pool. The system dissipates decay heat from stored fuel to the component cooling water system. The system is designed to handle the decay heat generated by one-third of a core in the pool, and the temperature of the spent fuel pool water under this condition will not exceed 120°F. If it is necessary to remove a complete core from the reactor while the spent fuel assemblies from the previous refueling remain in the pool, the system can maintain a pool water temperature at or below 150°F.

We have reviewed the design of the spent fuel pool cooling and cleanup system, and conclude that it meets the intent of the positions set forth in Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," and AEC General Design Criterion No. 61. Therefore, the design is acceptable.

9.2.3 Fuel Handling System

For the Diablo Canyon design, the cask loading area is in the spent fuel pool. During cask handling operations, unacceptable damage resulting from a spent fuel cask drop will be prevented by limiting the

travel of the spent fuel cask over areas that contain no safety related equipment or stored fuel. The travel of the cask bridge crane is limited by mechanical stops and limit switches. The subject of cask movement over spent fuel stored in the pool has been discussed with the applicant. PG&E has tentatively agreed that spent fuel will not be stored in the spent fuel pool in locations where it could be struck by a dropped cask, assuming the worst tipped position for the dropped cask. However, final resolution of this item is still pending, and will be reported on in a supplement to this Safety Evaluation Report.

We have considered in our review the design safety features of all the fuel handling equipment, and judge that the system design meets the intent of the positions stated in Regulatory Guide 1.13 and the requirements set forth in General Design Criterion No. 62. We conclude, subject to favorable resolution of the dropped cask item, that the design of the fuel handling system is acceptable.

9.3 Water Systems

9.3.1 Auxiliary Saltwater System

The auxiliary saltwater system provides cooling water from the Pacific Ocean to the component cooling water heat exchangers. For each of the Diablo Canyon Units, two separate pump compartments are located in the intake structure which is located on the Pacific Ocean. Each of the pump compartments contains a full capacity pump. Each unit is

provided with two pumps, which are headered into two separate trains. The double valved, normally open interconnection between the two trains is provided with remote operating capability. A normally closed motor-operated valve provides separation between the Unit 1 and Unit 2 auxiliary saltwater discharge headers.

The entire auxiliary saltwater system, including the intake structure, is designed to withstand the effects of the safe shutdown earthquake. Furthermore, the system is designed such that a single failure of any component of the system, or the onsite power system, will not prevent a safe shutdown of the plant.

We have reviewed the design of the auxiliary saltwater system and have concluded that it meets the requirements of AEC General Design Criterion No. 44, and is acceptable.

9.3.2 Component Cooling Water System

The component cooling water system is designed to remove residual and sensible heat from the reactor coolant system via the residual heat removal system during plant shutdown, to cool the spent fuel pool water, to cool the letdown flow to the chemical and volume control system during power operation, and to provide cooling to dissipate waste heat from various primary plant components. The system design provides means for detection of radioactivity entering the system from the reactor coolant system and associated auxiliary systems, and includes provisions for isolation of system components as required.

During normal full power operation, two of the three component cooling water pumps and one of two component cooling water heat exchangers accommodate the heat removal loads. The standby pump and heat exchanger provide backup during normal operation. Both component cooling water heat exchangers provide removal of residual and sensible heat during a normal plant shutdown. Failure of one of these heat exchangers increases the time required for shutdown, but does not affect the safe shutdown of the plant.

In the event of a loss of coolant accident, one pump and one heat exchanger are capable of fulfilling system requirements. The system design provides three main cooling headers, two isolable headers that supply cooling water to essential safety equipment, and one header which supplies cooling water to the other plant auxiliaries. Under this arrangement, long-term cooling of the engineered safety features under accident conditions is assured, assuming either active component failures or the development of excessive leakage in one header of the component cooling water system. Cooling water for the component cooling water heat exchangers is supplied from the auxiliary saltwater system, thereby assuring a continuous source of cooling under all conditions.

The component cooling water system is a Seismic Category I system which is required for post-accident removal of decay heat from the reactor. As such, the system is designed to meet the single failure criteria with two completely independent, parallel trains available,

each containing one pump and one heat exchanger. The applicant presented in the FSAR the results of a failure mode and effects analysis of the system pumps, heat exchangers, and valves; we found the results of this analysis to be acceptable.

We have reviewed the design of the component cooling water system and have concluded that it meets the requirements of AEC General Design Criterion No. 44, and is acceptable.

9.3.3 Makeup Water System

The makeup water system utilizes distilled seawater to supply 150 gpm of distillate to storage tanks, makeup demineralizers, and reservoirs. This system is designed to provide a source of water for makeup to the reactor coolant loop, supply and makeup for the fire protection system and the spent fuel pool, and supply for the component cooling water system. The makeup water system, from the condensate storage tanks to the auxiliary feedwater pumps, and from the condensate storage tanks through the makeup transfer pumps to the component cooling water system, is Seismic Category I in order to meet the design requirements with respect to safety related systems.

We have reviewed the design of the makeup water system and conclude that it is acceptable.

9.3.4 Ultimate Heat Sink

The ultimate heat sink consists of the Pacific Ocean. This body of water, together with the intake structure and the discharge system

in conjunction with the auxiliary saltwater system, provides a means of supplying cooling water for reactor equipment to use as a heat sink. The intake structure is located directly on a small inlet cove of the ocean, thereby assuring the source of cooling water (see Figure 2.2 of this report). With regard to low water level conditions in the cove, the most severe oceanographic phenomenon would be a tsunami (see Section 2.4.6 of the FSAR). The expected downsurge from the tsunami during short periods of time would be to 9 feet below MLLW (mean lower low water). The arrangement of the intake channel and the design of the auxiliary saltwater pumps allows operation down to 17.4 feet below MLLW. Thus, the operation of the Seismic Category 1 auxiliary saltwater pumps is assured during extreme tsunami drawdown. For reference, MLLW equals MSL (mean sea level) minus 2.6 feet. MSL is ground elevation zero.

Based on our evaluation of the ultimate heat sink, we conclude that the design meets the positions set forth in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and is acceptable.

9.4 Process Auxiliaries

9.4.1 Chemical and Volume Control System

The chemical and volume control system (CVCS) is designed to adjust the concentration of chemical neutron absorber (boron) in the reactor coolant for reactivity control, maintain the proper water inventory and concentration of corrosion inhibiting chemicals in the reactor coolant system, provide required seal water injection to the

reactor coolant pump seals, and remove corrosion products and fission products from the reactor coolant. Two separate and independent flow paths are available for reactor coolant boration, i.e., the charging line and the reactor coolant pump seal injection. In the event of loss of offsite power, the safety (boration) function of the system would be maintained, in that power to the charging pumps and associated valves would be available from the diesel generators. To evaluate the safety of the system, failures or malfunctions were assumed concurrent with a loss-of-coolant accident, and the consequences evaluated. The results of the failure analyses were found to be acceptable.

On the basis of the similarity of the design of the Diablo Canyon chemical and volume control system to that for previously reviewed and approved plants, such as Zion Units 1 and 2, we have concluded that the system is acceptable.

9.5 Air Conditioning, Heating, Cooling, and Ventilation Systems

9.5.1 Control Area

The control area air conditioning, heating, and ventilation systems are designed to provide a controlled environment for the control room, the computer room, and the instrument safety-feature rooms for each Unit. These systems are designed to Seismic Category I requirements, and are required in order to assume continuous occupancy of the control room under normal and accident conditions.

As a result of our review, the applicant has added chlorine detectors in the control room supply duct, with local and remote alarms (see Section 6.4 of this report). The remote alarm will be used for automatic initiation of an internal recirculation operational mode, the same as would be initiated for a high activity alarm in the supply duct. The systems also have features for protection from smoke generated either inside or outside the control room area. The systems have two full capacity units of equipment for each primary element. The source of power for each electrically powered primary unit of equipment is from a vital bus.

We have reviewed the design of the control area air conditioning, heating, and ventilation systems, and conclude that they are acceptable.

9.5.2 Auxiliary Building (Excluding the Fuel Handling Area)

The auxiliary building heating and ventilation system has the primary function of maintaining the temperature of the engineered safety feature pump motors within acceptable limits during their operation. The secondary function of this system is to provide heating and ventilation to the auxiliary building.

The system is designed, built and installed to Seismic Category I requirements. The power for the fans and initiating (logic) circuitry is taken from vital buses. All dampers are designed to assume the position required for emergency operation upon failure of damper motive power. All primary active components of the system, including initiation circuitry, are redundant.

We have reviewed the system design with respect to its effects on the operation of equipment essential for safe shutdown of the reactor, and conclude that it is acceptable.

9.5.3 Fuel Handling Area of the Auxiliary Building

The principal function of the fuel handling area heating and ventilation system is to sweep radiolytic gases from the surface of the spent fuel pool, and to treat the exhaust air in order to remove most of these gases. The purpose of treating the exhaust air is to reduce the offsite dose to acceptable levels in the event of a fuel handling accident. The system is designed and built to Seismic Category I requirements.

The fuel handling area for each of the Diablo Canyon Units is physically isolated from the rest of the auxiliary building. The heating and ventilation system consists of redundant supply fans, redundant exhaust fans, and redundant HEPA and charcoal filter banks. A third full capacity exhaust fan and HEPA filter bank train is provided for normal operation, and is either automatically or manually prevented from operating when exhaust cleanup through the charcoal filters is required.

We have reviewed the design of the fuel handling area heating and ventilation system, and conclude that it is acceptable.

9.5.4 Intake Structure (Auxiliary Saltwater Pump Compartments)

Units 1 and 2 are each provided with two redundant auxiliary saltwater pumps. Each pump is installed in a separate watertight compartment in the

intake structure.

Proper ventilation of these compartments is assured by providing each compartment with a separate ventilation system consisting of intake louvers, an exhaust fan and an exhaust duct. The power sources for these fans are supplied by ESF buses. The function of this system is to maintain the temperature of the auxiliary saltwater pump motors within acceptable limits during their operation. The system is designed and built to Seismic Category I requirements. Minimum saltwater pumping requirements are met, assuming a single failure in the ventilation system.

We have reviewed the design of the ventilation system for these compartments, and conclude that it is acceptable.

9.5.5 Diesel Generator Compartments

The ventilation systems for the diesel generator compartments have the function of maintaining the air temperature of the compartments within acceptable limits during diesel operation and providing cooling air for the radiator of the diesel generator closed loop jacket water system.

Each diesel generator compartment has its own Seismic Category I ventilation system. Ventilation of the compartments is assured when the diesels are operating, since the ventilation for each compartment is provided by the same direct engine-driven fan which provides cooling air to the radiator. No special provisions for heating the diesel generator are required since the diesel jacket water and lubricating oil

are kept warm by thermostatically controlled heaters during periods when the diesel generators are not operating.

We have reviewed the design of the diesel generator compartment ventilation system, and conclude that it is acceptable.

.6 Other Auxiliary Systems

.6.1 Fire Protection System

The fire protection system has been designed to: (1) provide automatic or manual fire extinguishing capability; (2) provide redundant detection equipment throughout the plant; (3) provide automatic suppression systems in area where hazardous materials are stored, when the malfunction of these systems will not interfere with the function of essential equipment; and (4) comply with the standards of the National Fire Protection Association.

Two reliable water supplies, a 300,000 gallon storage tank and a 4.5 million gallon raw water storage reservoir, are provided as sources of water for fire protection. Water is piped to all levels of all facility buildings (both units), with the supply lines sectionalized by valves for isolation in the event of damage to any section of the line. Water spray systems are provided at high hazard locations, with hand hoses and CO_2 devices at all other points. In order to minimize the potential of fire throughout the plant, non-combustible and heat resistant materials have been used wherever practicable.

On the basis of our review, we have concluded that the fire protection system conforms with the intent of the requirements of AEC General Design Criterion No. 3 and, therefore, is acceptable.

9.6.2 Diesel Generator Auxiliary Systems

Each diesel generator has two independent air starting systems powered by separate d-c power sources. The original design provided automatic switching of d-c power sources if the primary source failed. We concluded that this was unacceptable since a single failure in the automatic switching could result in the loss of two independent d-c power sources. The applicant modified the design providing for manual switching of the standby d-c power source for starting. We have concluded that the modified design meets our requirements and is acceptable.

Each air starting system consists of an air receiver which provides 45 seconds of continuous engine cranking, an air compressor, and two air starting motors. The system is Seismic Category I. Diesel combustion air is taken in through Seismic Category I filters from the west wall of the building, and the engine exhaust is directed out the north side of the building; this assures that the exhaust will be removed without diluting the combustion air. The lubricating oil system for each engine is entirely contained on the engine's base plate. The oil is cooled in the jacket cooling water heat exchanger. This closed cooling water system is in turn cooled by the engine air radiator. Thermostatically controlled inversion heaters keep the jacket water and lubricating oil warm for fast starting when the engine is in a shutdown condition. In

the event of a high energy line break in the turbine building, it is possible that the air flow from the turbine building to the generator compartments could be diluted slightly to a steam air mixture for a short period of time. This would not affect the engine combustion air, and would not be expected to affect operation of the engine generator since the engines that will be in use at Diablo Canyon are routinely subjected to a high humidity environment.

9.6.3 Diesel Generator Fuel Oil System

A 40,000 gallon fuel oil storage tank for each unit is the source of fuel oil supply for the diesel generators. Each tank can supply one diesel with enough oil to run it for seven days at full load. Two diesel fuel oil transfer pumps, capable of taking suction from either storage tank, feed a 550-gallon day tank on each of the five diesel generators. This 550-gallon capacity is sufficient for two and one-half hours of full load operation. The day tank is provided with level indication and overflows. The entire diesel engine fuel oil system is designed to Seismic Category I requirements. During loss of offsite power conditions, only one diesel generator is required for each unit.

We have reviewed the design of the diesel generator fuel oil system, and conclude that it is acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system for Units 1 and 2 is of conventional design, similar to those of previously approved plants. Each system is designed to remove heat energy from the reactor coolant in four steam generators and convert it to electrical energy by a turbine driven generator. The condenser transfers unusable heat in the cycle to the condenser cooling water. The entire system is designed for the maximum expected energy from the nuclear steam supply system. Upon loss of full load, the system is capable of dissipating the energy in the reactor coolant through bypass valves to the condenser or through power operated relief valves, dump valves, and safety valves to the atmosphere.

10.2 Turbine-Generator

The main turbine-generator and auxiliary systems are designed for steam flows corresponding to 3496 and 3592 MWt for Units 1 and 2, respectively. The Unit 2 turbine-generator has a higher power rating, thereby causing the subsequent higher rating of the Unit 2 NSSS. The turbine generator consists of a four casing, tandem-compound, six flow exhaust unit with a design speed of 1800 rpm. The turbine has one double-flow high-pressure element in tandem with three double-flow low-pressure elements. Moisture separation and reheating of the steam is provided

between the high- and low-pressure turbines by six horizontal-axis, two stage reheat cylindrical-shell combined moisture-separator-reheater assemblies.

The turbine electro-hydraulic control system will control the speed of the turbine by modulating the turbine inlet steam control valves to control the steam flow to the turbine. The control system regulates turbine speed prior to the time that the generator is synchronized, and controls unit output when the generator is connected to the grid. The turbine control system will be designed to automatically trip the turbine under the following conditions: turbine overspeed, low condenser vacuum, thrust bearing failure, low bearing oil pressure, high-high steam generator water level or safety injection, generator electrical trips, and reactor trip. The turbine overspeed trip system consists of a mechanical trip which functions at 110 percent overspeed, and an electrical solenoid trip which is actuated at 111.5 percent overspeed.

We have reviewed the turbine generator control systems and protective devices, and conclude that they are acceptable.

10.3 Main Steam Supply System

The steam generated in each of four steam generators will be routed to the turbine by means of the main steam lines. Each steam line contains a flow restrictor to limit maximum flow and the resulting thrust loading caused by a steam line rupture. Each line will contain one main steam isolation valve (stop valve). The four main steam lines

will join together in a pressure and temperature equalization header from which the steam is normally routed to the high pressure turbine.

Steam dissipation capability in the event of a turbine and/or reactor trip will be as follows:

(1) The turbine bypass system will have the capacity of dumping 40 percent of full flow steam to the main condenser. This, in conjunction with a 10 percent step load change capability of the nuclear steam supply system, permits a 50 percent turbine generator load rejection without a reactor trip, turbine trip, or safety valve actuation;

(2) The power operated relief valves will have the capacity of dumping 35 percent of full flow steam to the atmosphere. These valves will be used for controlled cooldown during loss of offsite power conditions;

(3) The steam generator safety valves, having a capacity in excess of 100 percent full steam flow, will open as a last resort. They will be set to relieve at higher pressures than the power relief valves.

The main steam isolation valves (back to back check valves) will be designed to provide positive isolation against forward or reverse steam flow. The main steam line isolation valves will close if there is evidence of a main steam line rupture, as indicated by signals of containment high pressure or high steam flow coincident

with low steam line pressure or low average temperature. Closing time will be 5 seconds or less. The valves can be operated from the control room. The applicant is presently performing an analysis of the ability of the main steam check valves to remain functional following a steam line break upstream of the valve. Our evaluation of this analysis will be reported on in a supplement to this Safety Evaluation Report.

We have reviewed the main steam system and isolation valve designs, together with the testing program, and we conclude, subject to favorable resolution of the item discussed above regarding integrity of the check valves, that these designs are acceptable.

10.4 Other Features

This section discusses subsystems of the steam and power conversion system that are used during the process of converting thermal energy to electrical energy. These include the main condenser, turbine bypass system, circulating water system, and the condensate and feedwater systems. Other subsystems of the steam and power conversion system that have been reviewed but not discussed in detail are the main condenser evacuation system, the turbine gland sealing system, and the condensate and feedwater chemical injection system. These system designs have been found to be similar to those of previously approved facilities, and are acceptable. The steam generator blowdown system is discussed in Section 11 of this report.

The main condenser has been designed to serve as a heat sink for the turbine exhaust steam, turbine bypass steam and other flows. The condenser hotwells are sized to provide adequate storage of water to allow for losses to the atmosphere and shrinkage on a full load trip. During operation, air is removed from the condenser by steam jet air ejectors and discharged to the plant vent. The discharged air is continuously monitored for radioactivity.

The turbine bypass system (TBS) discharges main steam directly to the condenser during the emergency condition of a sudden load rejection by the turbine generator or turbine trip, and during plant startup and shutdown. The turbine bypass system has been designed for a total steam flow capacity equivalent to 40 percent of the full-load main steam flow. There are twelve power relief valves which take steam from the dump header; these valves discharge into spray distribution headers in the condenser. Four of these twelve valves are used during cooldown of the reactor.

The circulating water system furnishes the main condenser with cooling water from the Pacific Ocean. For optimum turbine-condenser performance, the circulating water system utilizes two pumps to supply 876,000 gpm of salt water to the condenser of each unit. The water passing through the condenser is returned to the Pacific Ocean via the discharge structure.

Because the circulating water system is not Seismic Category I, the staff has considered the potential effects of flooding due to a break in this system. In particular, we have reviewed the consequences of the rupture of a circulating water line expansion joint. In order to preclude flooding of safety related equipment in the turbine building due to the rupture of this joint, the applicant has installed an expansion joint sleeve around each expansion joint. This sleeve provides an essentially watertight barrier. Weep holes on each expansion joint sleeve are provided for indication of expansion joint failure. We have concluded that by the use of these sleeves, an expansion joint failure will not adversely effect the safe shutdown of the reactor.

With regard to the remaining exposed portions of the circulating water pipe, we have requested that the applicant supply design modifications to the turbine building to ensure that rupture of this pipe would not impair safe shutdown of the reactor. Specifically, in order to prevent water from entering the compartments containing the diesel generators, the applicant has proposed a two foot high door separating the entrance to the emergency generator hallway from the main turbine building floor. The staff is currently reviewing this proposed design modification. Resolution of this item will be discussed in a supplement to this Safety Evaluation Report.

The condensate and feedwater system has been designed to process the condensate to maintain the required quality of feedwater and provide

the required amount of feedwater to the steam generators at the associated feedwater temperature and margin of flow to accommodate all anticipated transient conditions.

From our review of the steam and power subsystems, and subject to favorable resolution of the item discussed above regarding modifications to the turbine building, we conclude that the design of these subsystems is acceptable.

10.5 Auxiliary Feedwater System

The auxiliary feedwater system supplies water to the steam generators for reactor decay heat removal if the normal feedwater sources are unavailable due to loss of offsite power or other malfunction. Each unit is equipped with one full capacity turbine-driven and two half capacity motor-driven auxiliary feedwater pumps. Steam for the turbine-driven pump is taken from two of the four main steam lines upstream of the steam generator isolation valves. Separate isolation valves are provided for these branch connections. The motor-driven pumps receive power from the vital buses.

The turbine-driven auxiliary feed pump, rated at 930 gpm and 3000 feet discharge head, and the motor-driven auxiliary feed pumps, rated at 490 gpm and 3000 feet discharge heat, take suction from the condensate storage tank. Each auxiliary feedwater pump discharges to all four of the steam generators. Feedwater flow is controlled from the control room by remotely operated flow control valves in the supply lines to each steam generator.

The turbine-driven pump automatically starts upon loss of offsite power by opening the turbine stop-start valve in the steam supply line. The motor-driven pumps start automatically during conditions of loss of offsite power or loss of the turbine-driven feed pump system. The pumps can be started from the control room. The system has been designed and built to Seismic Category I requirements.

The condensate storage tank, from which the auxiliary feed pumps take suction, is also Seismic Category I. The tank has a capacity of 425,000 gallons, of which 170,000 gallons is required for the auxiliary feedwater system during plant cooldown. The tank is adequately protected from the effects of flooding, and has a tornado resisting capability (combined wind and missile) for winds up to 150 mph. Missile damage to the tank below the 170,000 gallon level could impair its safety related function. The tank has sufficient capacity to cool the reactor coolant to 350°F and reduce system pressure to 350 psig. The 4.5 million gallon raw water reservoir serves as a backup source of water.

We have reviewed the auxiliary feedwater system capability in accordance with the criteria set forth in A. Giambusso's letter of December 1972 with respect to high energy piping system breaks outside containment (see Section 3.6 of this report). Based on this review, we have determined that if a high energy line fails outside of containment,

and assuming a concurrent single active component failure in the auxiliary feedwater system, that the minimum feedwater flow from one of the motor-driven pumps (490 gpm at 3000 feet discharge head) is sufficient for removal of reactor decay heat.

We have reviewed the auxiliary feedwater system and the condensate storage facilities, and conclude that their designs are acceptable.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The radioactive waste systems, which will be shared by the two units, consist of the liquid, gaseous, and solid waste systems.

The liquid waste system will process waste liquid streams such as equipment drains, leakage, blowdown demineralizer regenerant waste, decontamination and laboratory waste liquids, and laundry and shower waste water. The treated liquid waste will be recycled for reuse if the plant waste balance requires makeup and if the water quality is adequate. The liquid waste system will utilize evaporation, demineralization, and filtration for removal of radioactive material, chemical impurities, and particulates.

Gaseous wastes will be generated during the operation of the plant as a result of degassing of the primary coolant, vents from equipment handling radioactive materials, and leakage from systems and components containing radioactive material. The gaseous waste system will remove radioactive materials from gaseous streams by filtration, adsorption, and holdup for radioactivity decay. The treated gas streams will be released to the environment.

Solid wastes will also be generated during plant operation. The wastes will consist of such contaminated material as clothing, evaporator bottoms, demineralizer resins, and discarded radioactive components and tools. Treatment will consist of solidification, packaging, and shipping to a licensed burial site.

BLANK PAGE

In the Final Environmental Statement (FES) for Diablo Canyon, Units 1 and 2, dated May 1973, the staff performed an evaluation to determine the quantities of radioactive materials that will be released in the liquid and gaseous plant effluents, and that will be shipped offsite as solid wastes for burial. In that evaluation, waste flows, waste activities, and equipment operating performance that are consistent with normal plant operation were considered, including anticipated operational occurrences over the life of the plant. The parameters used in the FES evaluation, along with their bases, are given in Appendix B to WASH-1258. Modified versions of the ORIGEN and STEFFEC Codes, which were the liquid and gaseous calculational models that were used, are discussed in Appendix C to WASH-1258. Our evaluation of the system decontamination factors, along with a listing of plant dependent parameters, is given in Table 3.5 of the FES.

11.2 Liquid Waste System

11.2.1 System Description

The liquid radwaste treatment system is designed to collect and process wastes based on the chemical purity, relative to the primary coolant, as determined by the origin of the waste in the plant. The boron recycle system (BRS) will process shim bleed and equipment drain waste, collected inside the reactor containment, by means of evaporation and demineralization. The liquid waste treatment system (LWTS) will process, by evaporation and demineralization, equipment drain wastes and

tank overflow wastes from components outside reactor containment, wastes from laboratories and sampling drains, and demineralizer regeneration solutions and mi .laneous low purity wastes which have been collected in floor drains and building sumps. The LWTS will also process detergent wastes and/or turbine building floor drain wastes should radiation measurements indicate higher than expected radioactivity levels. These wastes will normally be filtered and released without treatment, after monitoring for radioactivity. The steam generator blowdown treatment system (SGBTS) will process blowdown wastes by mixed-bed demineralization. The BRS and LWTS are shared between Units 1 and 2, while the SGBTS is a separate system for each unit. The principal components making up each of these systems, along with their capacities, are listed in Table 11.1.

11.2.2 System Evaluation

In our evaluation of the liquid radwaste system we have considered the following criteria: (1) the capability of the system to reduce radioactive releases to "as low as practicable" levels based on expected radwaste inputs over the life of the plant; (2) the capability of the system to maintain releases below the limits in Appendix B of 10 CFR Part 20 (see Table 2, Column II for periods of fission product leakage at design levels from the fuel); (3) the capability of the system to meet the processing demands of the plant during anticipated operational occurrences; (4) the quality group classification and seismic category

applied to the system design; and (5) the design features incorporated to preclude uncontrolled releases of radioactive materials due to tank overflows. The process and effluent monitoring design capabilities are considered in Section 11.4 of this report.

Our evaluation of the liquid radwaste treatment system for normal operation is given in the Final Environmental Statement for Diablo Canyon. In the FES we have determined that the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 5 Ci/yr/reactor, excluding tritium and dissolved gases, and 350 Ci/yr/reactor for tritium. An isotopic listing of our calculated liquid source term is given in Table 3.6 of the FES. Based on that evaluation, we have found that the release of radioactive materials in liquid effluents will not result in whole body or critical organ doses in excess of 5 mrem/yr at or beyond the site boundary, and that radioactive materials released in liquid effluents, exclusive of tritium and dissolved gases, will not exceed 5 Ci/yr/reactor. We have reviewed the effects of reactor operation with one percent of the operating fission product inventory in the core being released to the primary coolant. We have determined that under these conditions, the concentrations of radioactive materials in liquid effluents will be a small fraction of the limits given in Appendix B of 10 CFR Part 20.

The design capacities of the BRS and LWTS evaporators are each 21,000 gallons per day (gpd). We calculate the average expected waste

flows to the BRS and the LWST to be 2880 and 3430 gpd respectively, for the combined wastes from both units. The differences between the expected flows and design capacities provide adequate reserve for processing surge flows. In addition, the design allows wastes to be processed interchangeably between the two systems in the event of equipment downtime. We conclude that the system capacities and designs are adequate for meeting the demands of the plant during anticipated operational occurrences.

The SGBTS is a separate system for each reactor. Each SGBTS consists of two mixed-bed demineralizers, capable of passing up to 65 gpm in series flow. The system design also allows for untreated blowdown discharges to be monitored. Automatic closure of an isolation valve terminates releases if a preset radiation level is reached. We calculate the average expected blowdown rate to be approximately 24,000 gpd/unit. The capacity of each SGBTS is greater than 86,000 gpd. We conclude, therefore, that the system capacity and design are adequate for meeting the demands of the plant during anticipated operational occurrences.

The liquid radwaste systems are located in a Seismic Category I structure. The failure of system components would not result in radionuclide concentrations in excess of the limits in 10 CFR Part 20 in the nearest potable water supply or at the nearest surface water supply. The quality group designations of the equipment in the liquid radwaste systems are listed in Table 11.1 The systems are designed to

preclude the uncontrolled release of radioactive materials due to overflows from indoor and outdoor tanks by providing level instrumentation which will alarm in the control room, and by means of curbs and Seismic Category I vaults to collect liquid spillage and retain it for processing. We consider these provisions to be capable of preventing the uncontrolled release of radioactive materials to the environment.

11.2.3 Liquid Waste System Evaluation Findings

The liquid radwaste system includes the equipment and instrumentation to control the release of radioactive materials in liquid effluents. The scope of our review included: (1) the capability of the system to reduce releases of radioactive materials in liquid effluents to "as low as practicable" levels in accordance with 10 CFR Parts 20 and 50.36a, considering normal operation and anticipated operational occurrences: (2) the design provisions incorporated to preclude uncontrolled releases of radioactive materials in liquids due to leakage or overflows in accordance with AEC General Design Criterion No. 60; and (3) the quality group classification and seismic design criteria. The review has included an evaluation of effluent releases based on the proposed treatment processes, and was based on information obtained from piping and instrumentation diagrams, schematic diagrams, and descriptive information in the FSAR. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the liquid radwaste system to the Commission's Regulations

and to applicable Regulatory Guides, as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed liquid radwaste system is acceptable.

11.3 Gaseous Waste System

11.3.1 System Description

The gaseous radwaste treatment system is designed to process wastes based on the origin of the wastes in the plant and their expected activity levels. The gaseous waste processing system (GWPS) will process gases stripped from the primary coolant boron recycle system gas stripper and miscellaneous tank cover gases by means of continuous recirculation through pressurized storage tanks (waste gas decay tank system). The GWPS is shared between Units 1 and 2. Radioactive gases from the main condenser steam air ejector will be released to the plant vent without specialized treatment. Radioactive gases vented from the SGBTS will be routed to the main condenser. When the untreated portion of the steam generator blowdown system is in operation, as discussed in Section 11.2.2 of this report, gases will be vented and the discharges monitored. Automatic closure of an isolation valve will terminate discharges if a preset radiation level is reached. Ventilation exhausts from the radwaste, fuel handling, and safeguards areas will be processed through HEPA (high efficiency particulate air) filters and charcoal adsorbers prior to release. The auxiliary building ventilation system normally exhausts to the plant vent through HEPA filters, but an alternate route

will permit passage through charcoal adsorbers. In addition, the containment building atmosphere will be recirculated through filters and charcoal adsorbers prior to purging. The turbine building ventilation exhausts will be released without treatment. The principal components in the GWPS, along with their capacities, are listed in Table 11.2.

11.3.2 System Evaluation

In our evaluation of the gaseous radwaste system we have considered the following criteria: (1) the capability of the system to reduce radioactive releases to "as low as practicable" levels based on expected gaseous waste inputs and radioactive leakage rates over the life of the plant; (2) the capability of the system to maintain releases below the limits in Appendix B of 10 CFR Part 20 (see Table 2, Column I for periods of fission product leakage at design levels from the fuel); (3) the capability of the system to meet the processing demands of the plant during anticipated operational occurrences; (4) the quality group classification and seismic category applied to the system design; and (5) the potential for gaseous releases due to hydrogen explosions. The process and effluent monitoring design capabilities are considered in Section 11.4 of this report.

Our evaluation of the gaseous radwaste treatment system for normal operation is given in the FES for Diablo Canyon. In the FES, we have determined that the proposed gaseous radwaste treatment systems will be

capable of reducing the release of radioactive materials in gaseous effluents to approximately 3700 Ci/yr/reactor of noble gases and 0.28 Ci/yr/reactor of iodine-131. An isotopic listing of our calculated gaseous source term is given in Table 3.7 of the FES. Based on that evaluation, we have found that the release of radioactive materials in gaseous effluents will not result in an annual air dose, at or beyond the site boundary, in excess of 10 mrad for gamma radiation and 20 mrad for beta radiation; the annual thyroid dose to an individual will not exceed 15 mrem, considering the location of the nearest cow (9.5 miles east of the reactors); and the annual quantity of iodine-131 released will not exceed 1 Ci for each reactor at the site. We have reviewed the effects of reactor operation with one percent of the operating fission product inventory in the core being released to the primary coolant. We have determined that under these conditions, the concentrations of radioactive materials in gaseous effluents will be a small fraction of the limits given in Appendix B of 10 CFR Part 20.

Operating with three 40 scfm compressors (one unit in continuous use and the others as backup), and six 705 ft^3 gas decay tanks (each of which is capable of being isolated from all the others), the system has adequate capacity to allow operation during periods of equipment downtime. We conclude that the system capacity and design are adequate for meeting the demands of the plant during both normal operation and anticipated operational occurrences.

Most of the gas entering the GWPS during normal operations will be cover gas displaced from the boron recycle holdup tanks as they fill with liquid. This gas will consist primarily of nitrogen and hydrogen. To prevent oxygen buildup in the system, the vent header is designed to operate at a slightly positive pressure. In addition, gas samples will be periodically drawn from the tanks discharging to the vent header, and from the decay tanks being filled. An alarm will be activated if the oxygen content of any sample exceeds two percent by volume (v/o). In this manner, the potential for explosive hydrogen/oxygen mixtures will be mitigated.

Gaseous wastes from the main condenser will not be treated. The system releases will be proportional to the rate of primary to secondary system leakage and the primry coolant activity. In the event of excessive primary to secondary leakage, the affected steam generator(s) will be isolated before radioactive material concentrations in main condenser offgas releases exceed the limits in the technical specifications.

The plant ventilation systems are designed to induce air flows from potentially less radioactively contaminated areas to areas having a greater potential for radioactive contamination. The ventilation system has adequate capacity to limit radioactive material concentrations in areas within the plant that are accessible during operation to below the limits in 10 CFR Part 20.

11.3.3 Gaseous Waste System Evaluation Findings

The gaseous radwaste system includes the equipment and instrumentation to control the release of radioactive materials in gaseous effluents. The scope of our review included: (1) the capability of the system to reduce releases of radioactive materials in gaseous effluents to "as low as practicable" levels in accordance with 10 CFR Parts 20 and 50.36a, considering normal operation and anticipated operational occurrences; (2) the design provisions incorporated to reduce the potential for hydrogen explosions; and (3) the quality group classification and seismic design criteria. The review has included an evaluation of effluent releases based on the proposed treatment processes, and has considered pathways due to process vents and leakage affecting building ventilation systems. The review was also based on information obtained from piping and instrumentation diagrams, schematic diagrams, and descriptive information in the FSAR. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the gaseous radwaste system to the Commission's Regulations and to applicable Regulatory Guides, as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed gaseous radwaste system is acceptable.

11.4 Process and Effluent Radiological Monitoring System

11.4.1 System Description

The process and effluent radiological monitoring system is designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs. Scintillation detectors will be used for particulate monitoring in gaseous effluents. Geiger-Mueller detectors will be used for monitoring liquids and for monitoring radioactive gases in vent effluents. Gaseous iodine will be collected on replaceable, impregnated charcoal adsorbers which will be continuously monitored by scintillation detectors while in use. Systems which are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and analyzed in the plant laboratory.

Table 11.3 indicates the proposed locations and types of continuous monitors. Monitors on effluent release lines will automatically terminate discharges should radiation levels exceed a predetermined value.

11.4.2 System Evaluation

In our evaluation of the process and effluent monitoring system we have considered the capability of the system to: (1) monitor all normal and potential pathways for release of radioactive materials to

the environment; (2) control the release of radioactive materials to the environment; and (3) monitor the performance of process equipment and detect radioactive material leakage between systems.

We have reviewed the locations and types of effluent and process monitoring provided. Based on the plant design and on the continuous monitoring and intermittent sampling locations, we have concluded that all normal and potential release pathways, excluding the turbine building vent, will be monitored. Due to the high potential for exfiltration from the turbine building, which is a relatively open structure, it is not practical to monitor the potential gaseous releases from the turbine building. The design includes provisions for automatically terminating effluent releases in the event radiation levels in discharge lines exceed a predetermined level. We have also determined that the sampling and monitoring provisions are adequate for detecting radioactive material leakage to normally uncontaminated systems, and for monitoring plant processes which affect radioactivity releases. On this basis, we consider that the monitoring and sampling provisions meet the requirements of AEC General Design Criteria Nos. 13, 60 and 64 and the guidelines set forth in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants."

11.4.3 Process and Effluent Radiological Monitoring Evaluation Findings

The provisions for process and effluent radiological monitoring include the instrumentation and controls for monitoring and controlling the releases of radioactive materials in plant effluents, and the monitoring of the level of radioactivity in process streams. The scope of our review included the provisions for monitoring and controlling the release of radioactive materials in plant effluents in accordance with AEC General Design Criteria Nos. 60 and 64 and Regulatory Guide 1.21, and for monitoring radioactivity levels within the plant in process streams in accordance with AEC General Design Criterion No. 13. The basis for acceptance in our review has been conformance of the applicant's design, design criteria, and design bases for the process and effluent monitoring systems to the Commission's Regulations, as set forth in the General Design Criteria, and to the applicable Regulatory Guide, as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed provisions for monitoring process and effluent streams are acceptable.

11.5 Solid Waste System

11.5.1 System Description

The solid radwaste treatment system is designed to collect and process wastes based on their physical form and need for solidification prior to packaging. "Wet" solid wastes, consisting of spent demineralizer resins, evaporator bottoms, filter sludges, and chemical drain tank

effluents, will be combined with a cement-vermiculite mixture to form a solid matrix and sealed in 55-gallon drums. Dry solid wastes, consisting of ventilation air filters, contaminated clothing and paper, and miscellaneous items such as tools and glassware, will be compacted into 55-gallon drums using an industrial baling machine. The solid waste system is shared between Units 1 and 2.

11.5.2 System Evaluation

In our evaluation of the solid radwaste treatment system we have considered: (1) the system design objectives in terms of expected types, volumes, and activities of wastes processed for shipment offsite; (2) the design capacities of system components, method of operation, and capability of meeting the demands of the plant due to anticipated operational occurrences; (3) waste packaging and conformance to applicable Federal packaging regulations; (4) provisions for controlling potentially radioactive airborne dusts during baling operations; (5) ic design and quality group classification; and (6) provisions for onsite storage prior to shipping.

Our evaluation of the solid radwaste treatment system for normal operation is given in the FES for Diablo Canyon. In the FES we determined that the expected solid waste volumes and activities shipped offsite from each unit will be 250 drums/yr of "wet" solid waste containing an average of 20 Ci/drum, and 500 drums/yr of "dry" solid waste containing less than 5 Ci total.

Drum filling operations will be controlled remotely from consoles located outside the drum fill area. Drumming operations will have interlock features to prevent opening of filling valves when a drum is not properly positioned in the filling station. In addition, the equipment is designed so that any spills will be collected in a drain pan and prevented from dripping on the floor. Baling of dry wastes will be carried out inside a closed dust shroud. Wastes will be packaged in 55-gallon steel drums that meet DOT requirements, and will be shipped to a licensed burial site in accordance with AEC and DOT regulations.

Storage facilities for up to 2000 drums of solid radioactive wastes are provided in the fuel handling area of Unit 2, plus a solid radwaste storage area for 28 drums (high level), 300 drums (intermediate level) and 300 boxes (low level) of packaged solid radioactive wastes. Based on our estimate of 750 drums/yr/reactor, we find the storage capacity adequate for meeting the demands of the plant.

The spent demineralizer resin storage tank is designed as Quality Group C. Waste transfer piping is designed to Quality Group D and non-Seismic Category I requirements. Since the quantity of radioactive materials in the piping will not have a significant potential for uncontrolled release to the environs, we consider this design to be acceptable.

11.5.3 Solid Waste System Evaluation Findings

The solid radwaste system includes the equipment and instrumentation for solidifying and packaging radioactive wastes prior to shipment offsite for burial. The scope of our review included: (1) capability of the system for processing the types and volumes of wastes expected during normal operation and anticipated operational occurrences, in accordance with AEC General Design Criterion No. 60; (2) the provisions for handling wastes with regard to the requirements of 10 CFR Parts 20 and 71, and 49 CFR Parts 170-178; and (3) the quality group classification and seismic design criteria. The review has included the provisions for onsite storage and for the control of airborne dusts during dry waste compaction, and was based on information obtained from piping and instrumentation diagrams, schematic diagrams, and descriptive information in the FSAR. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the solid radwaste system to the Commission's Regulations and to applicable Regulatory Guides. as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed solid radwaste system is acceptable.

Table 11.1

Design Parameters Of Principal Components Considered In Liquid Radwaste Evaluation

System	Component	Number	Capacity, ea.	Quality Group
BRS	Recycle Holdup Tanks	5	83,200 gal.	C
BRS	Evaporator Feed Demin.	4	30 gpm	C
BRS	Evaporator	1	15 gpm	C
BRS	Evap. Cond. Demin.	2	30 gpm	C
LWTS	Chemical Drain Tank	1	1000 gal.	D
LWTS	Regenerant Receiver Tanks	2	15,000 gal.	D
LWTS	Equipment Drain Receiver Tanks	2	15,000 gal.	D
LWTS	Aux. Bldg. Sump	1	3600 gal.	D
LWTS	Floor Drain Receiver Tanks	2	15,000 gal.	D
LWTS	Misc. Equip. Drain Tank	1	2700 gal.	[illegible]
LWTS	Waste Condensate Tanks	2	15,000 gal.	
LWTS	Waste Evaporator	1	15 gpm	
LWTS	Mixed Bed Demin.	1	50 gpm	C
LWTS	Laundry and Hot Shower Analysis Tank	2	1000 gal.	D
LWTS	Waste Cor. Holding Tank	1	2000 gal.	D
SGBTS	Mixed-Bed Demineralizers	2	65 gpm	D
SGBTS	Steam Generator Flash Tank	2	200 gal.	D
SGBTS	Steam Generator Blowdown Tank	2	4360 gal.	D
Misc.	Spent Resin Storage Tank	2	200 ft^3	C

TABLE 11.2

Design Parameters Of Principal Components
Considered In Gaseous Radwaste Evaluation

System	Component	Number	Capacity, Ea.	Quality Group
WGPS	Compressor	3	40 scfm	C
WGPS	Decay Tanks	6	705 ft^3	C
WGPS	Surge Tanks	2	14 ft^3	C

Table 11.3

Process And Effluent Monitoring

Steam Monitored		Detector Type
Containment Purge exhaust	(particulate and iodine)	γ Scintillation
	(gas)	Geiger-Mueller (G-M)
Plant Vent (particulate)	(2 channels)	γ Scintillation
(gas)	(2 channels)	G-M
Residual Heat Removal Exh. (particulate)		γ Scintillation
Liquid Radwaste Release		γ Scintillation
Gas Decay Tank Discharge (gas)		G-M
Control Room Air (particulate)		γ Scintillation
Equipment Drain Receiver Recirculation		γ Scintillation
Condenser Air Ejector (gas)		G-M
Component Cooling Heat Exchanger Outlet (2 channels)		γ Scintillation
Steam Generator - secondary side liquid phase		γ Scintillation
Steam Generator Blowdown Tank Vent (gas)		G-M

12.0 RADIATION PROTECTION

12.1 Shielding

Shielding of a nuclear power facility for radiological protection during normal operation has two objectives: (1) to ensure that radiation limits to operating personnel and the general public, as set forth in 10 CFR Parts 20 and 50, are upheld; and (2) to ensure that radiation exposures to operating personnel during refueling, maintenance and inspections are maintained as low as practicable (ALAP), based on applicable provisions of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable [Nuclear Reactors]." We have reviewed the Diablo Canyon shielding design to determine if the design fulfills the generic objectives described above.

Radiation exposures at Diablo Canyon are intended to be maintained ALAP by classifying all plant areas into radiation zones based on expected frequency and duration of occupancy. Thus, the design of the radiation shield was carried out with consideration of the dose rate criterion for each zone. Shielding analysis was made with accepted computer codes. Consistent with the design, the applicant addressed the steps he took to assure that low dose rate zones were not likely to be compromised by inadvertent increases in radiation levels. Hence, pipes carrying radioactive liquids including piping penetrations, ducts, reach rods, tube withdrawal spaces, etc. were designed to be located in properly shielded compartments. When the tanks within compartments

contain significant quantities of radioactivity, they are shielded from each other. Thus, each component or tank within a compartment is isolated to allow for maintenance, inspection and some non-routine operations with ALAP radiation interferences from other components (or tanks). Movable shielding and means for its utilization will be available for use where permanent shielding is impractical.

Based on the above design and operating philosophy of the applicant, we conclude that adequate consideration has been given to the shielding of facilities and components to keep exposures to operating personnel and the public within the applicable limits of 10 CFR Part 20, and to reduce unnecessary exposure during normal operation of the plant, as described by Regulatory Guide 8.8. The effectiveness of the shield will, however, be evaluated by means of radiation surveys during initial startup and full power operation.

12.2 Ventilation

The ventilation systems for the Diablo Canyon Units are designed to provide a suitable radiological environment for personnel and equipment. The path of the ventilation air is from areas of low radioactivity toward areas of higher activity to ensure contamination control. Various compartments throughout the plant are provided with roughing and high efficiency particulate air (HEPA) filter banks, with charcoal filters added at selected locations, to preclude a buildup of airborne contamination. The ventilation system is also designed to limit doses at the

site boundary to within as low as practicable guidelines of proposed Appendix I of 10 CFR Part 50. The design criteria of the systems, the description of the systems, and the operating procedures discussed in the FSAR provide reasonable assurance that adequate consideration has been given to ventilation design for protection of in-plant personnel from airborne radioactivity hazards.

12.3 Health Physics Program

The basic objectives of an in-plant nuclear power facility health physics program are to limit radiation exposures to personnel to as low as practicable, and to comply with the appropriate regulations in 10 CFR Part 20. Educating the individual on radiation control standards and procedures to ensure achievement of these objectives is the responsibility of the health physics personnel at the Diablo Canyon Plant.

Company policy for radiation protection at the Diablo Canyon Units is carried out based on appropriate AEC regulations. Consistent with this policy, programs and procedures will be adapted which are consistent with Regulatory Guide 8.8. Among these are personnel dosimetry by film badges; respiratory protection including a respiratory fitting program; personnel protective clothing and personnel decontamination procedures; signs, tags, ropes and other access control measures to preclude unauthorized entry into high radiation areas; special work permits and procedures; testing and calibrating monitoring instrumentation; and maintenance of radiological reports and records.

Monitoring instrumentation and counting room equipment will be operated by the health physics staff. Portable radiological survey instrument inventory is satisfactory and of state-of-the art quality. Self-reading dosimeters are maintained by this group for recording daily exposures on exposure estimate cards. A routine bioassay program consisting of urinalysis and whole body counting will be performed by an outside contractor, and will provide supporting data on the effectiveness of the air monitoring program.

We conclude that the applicant's program for inplant radiation safety, as reflected by the health physics program and the concern of management for radiation control standards, as stated in the FSAR, is adequate to limit occupational exposures to within the limits set forth in 10 CFR Part 20 and Regulatory Guide 8.8.

12.4 Radioactive Materials Safety

The personnel qualifications, facilities, equipment and procedures for handling the byproduct, source and special nuclear material sources utilized for reactor startup and equipment calibration were reviewed. Based on the information provided in the FSAR and amendments, we conclude that there is reasonable assurance that these sources will be stored, handled, and used in a manner to meet the applicable radiation protection provisions of 10 CFR Parts 20 and 30.

12.5 Area Monitoring

The radiological monitoring system is designed to continuously

measure the radiation levels at fifteen selected locations within the plant. Each instrument of the system will have a local alarm and readout feature at the fixed location and in the control room. Dose rate levels will be recorded in the control room.

Airborne radioactivity monitoring will be performed by eleven fixed gas and particulate monitors located throughout the plant. These are supplemented by particulate and iodine-131 mobile continuous air monitors (CAM's) which will be used in specific locations, such as the fuel handling and radwaste areas, during operations that may cause airborne radioactivity. A routine grab sampling program will also be maintained as part of the air monitoring program.

We conclude that the area monitoring program, with its concomitant instrument calibration techniques, will provide satisfactory radiological protection to in-plant personnel.

13.0 CONDUCT OF OPERATIONS

13.1 Organization and Qualifications

The plant staff proposed by the applicant for one unit operation consists of 62 full-time employees, in addition to security personnel. The plant staff functions in the following groups: operations, maintenance and technical support The staff will expand to 80 employees onsite for two unit operation. The plant superintendent is responsible for all onsite activities in connection with the safe operation and maintenance of the plant. He reports to the manager of steam generation, who in turn reports to the vice president - electric operations.

The supervisor of operations directs day-to-day operation of the plant and is responsible to the plant superintendent for the operation of the facility. Normal shift operation is under the direct control of the shift foreman who is directly responsible to the supervisor of operations. The normal shift complement for single unit operation will consist of six employees, including a licensed senior reactor operator and three licensed reactor operators. For two unit operation, the applicant originally proposed a crew of eight, including one senior reactor operator. Such a shift crew is not in accordance with the Regulatory staff position which states that the number of licensed senior reactor operators onsite should not be less than the number of reactors that are operating. The applicant has now proposed that the crew staff during Unit 2 startup will number nine, and during commercial operation

BLANK PAGE

of both units, eight, including two licensed senior reactor operators and three licensed reactor operators. These shift crew compositions are in accord with the established staff position.

Normal plant maintenance is accomplished by a maintenance staff of 24, under the direction of the supervisor of maintenance. Maintenance personnel from the applicant's other thermal plants may be used to supplement plant forces as necessary for major repairs. The technical support section will have 14 personnel under the power plant engineer, who is responsible to the plant superintendent. The technical supp[illegible] provides services in the areas of reactor engineering, chemistry, [illegible] chemistry, radiation protection, reactor plant performance, and instrument and control systems. Qualifications for plant employees will meet the criteria set forth in ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel." The key management, supervisory and technical positions in the plant will initially be filled by employees whose experience in the nuclear power field ranges from eight to sixteen years.

The manager of steam generation functions as the engineer-in-charge (as defined in ANSI N18.1) of technical support activities for the plant staff. He has a technical staff of 14 in addition to the resources of technical specialists from other company departments. All of the engineering and scientific disciplines necessary to provide operational technical support are available within the applicant's organization so that PG&E does not plan for outside consultants to have assigned

responsibility for any specific area of technical support. However, consultants will be used to assist on special problems.

13.2 Training Program

The applicant has arranged a training program for operating personnel which was tailored to meet the needs of each man with respect to his previous background and job responsibilities. The program included a series of lectures provided by the nuclear steam system supplier covering the function, design description, control and instrumentation, normal and abnormal operation, and maintenance of the pressurized water reactor system of the Diablo Canyon reactors. Key individuals in the initial plant organization were assigned to an operating PWR to observe and participate in operations, with the length of assignment lasting about a month. Candidates for cold AEC licenses will attend a two-week training program at the reactor simulator of the nuclear steam supplier. Selected members of the plant technical support groups have completed formal training specifically oriented to their assigned responsibilities.

The applicant has described his proposed operator requalification program which will be placed into effect within three months after issuance of the Unit 1 operating license. The program includes a lecture series; on-the-job training with plant control manipulation; review of all design, procedure and license changes; semi-annual review of all abnormal and emergency procedures, and both written and oral examinations. We have reviewed the information submitted by the applicant, and have concluded that certain revisions must be made in the operator requalification program in order to meet the requirements of Section 50.54(i-1) of 10 CFR Part 50

and Appendix A of 10 CFR Part 55. Resolution of this item will be discussed in a supplement to this Safety Evaluation Report.

Subject to favorable resolution of the operator requalification program, we conclude that the proposed organization, the training and retraining, and the qualifications of the Diablo Canyon staff are adequate to provide acceptable staff and technical support for safe operation of the plant.

13.3 Emergency Planning

The emergency plan submitted by the applicant for Diablo Canyon Units 1 and 2 includes California's radiological emergency plan of the Bureau of Radiological Health of the Department of Public Health, dated January 1971. The plan also includes the San Luis Obispo County Sheriff's Department interim evacuation plan, dated June 1974, which contains specific instructions covering evacuation of people in the environs of the Diablo Canyon Plant.

In the event of an emergency, the normal operating crew is qualified to and responsible for making an initial evaluation of the incident, performing any immediate operations which are necessary, and placing appropriate portions of the emergency plan into effect. The senior member of the normal shift operating crew assumes the position of emergency coordinator until a senior member of plant staff arrives.

The emergency plan includes provisions for primary and alternate emergency control centers, notification of offsite state and federal agencies with responsibilities during an emergency, an emergency communications network, a description of onsite first aid and decontamination

facilities, and provisions for emergency transportation. The Sierra Vista Hospital in San Luis Obispo has agreed to accept and treat contaminated patients. The plan describes a spectrum of accidents with specific action levels for protective measures. The plan provides for annual training drills including checks of communications with local agencies.

We have reviewed the emergency planning program for the Diablo Canyon Plant, and we find that the program conforms with Appendix E of 10 CFR Part 50, and is acceptable.

13.4 Safety Review and Audit

The safety review and audit functions for the Diablo Canyon Plant will be performed by three separate and independent groups - the Plant Staff Review Committee, the General Office Nuclear Plant Review and Audit Committee, and the President's Nuclear Advisory Committee. The Plant Staff Review Committee is composed of senior plant staff personnel. It is advisory to the plant superintendent and will review all proposed tests and design modifications, and operating, maintenance and test procedures and changes thereto involving safety related aspects of safety related equipment. The General Office Nuclear Plant Review and Audit Committee is composed of individuals without line responsibility for the reactors. It provides reviews of audits, actions and practices bearing on nuclear safety, and initiates audits independent of the personnel directly responsible for the activity being audited. The President's Nuclear Advisory Committee is composed of four senior

managers in the PG&E organization. Its primary function is to examine and report to the president on the activities of the General Office Nuclear Plant Review and Audit Committee. The charters of both independent review and audit committees have been designed to conform to the guidance in ANSI N18.7-1972, "Standard for Administrative Controls for Nuclear Power Plants."

We conclude that the provisions for review and audit of plant operations are acceptable.

13.5 Plant Procedures

All safety related operations are to be performed in accordance with written and approved operating and emergency procedures. These procedures are incorporated into a Diablo Canyon Plant Manual, and include administrative, operating, emergency, maintenance, surveillance and test, radiation protection, chemical and radiochemical, and security procedures. The procedures conform to the guidance in Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," and are based on the requirements and recommendations of ANSI N18.7-1972.

We conclude that the provisions for preparation, review, approval, and use of written procedures are satisfactory.

13.6 Industrial Security

The applicant's initial proposal for industrial security for the Diablo Canyon Nuclear Plant was considered by the staff to be deficient in certain areas. Revisions to the security plan were submitted and

reviewed by the staff. In particular, the applicant has agreed to the arming of plant security guards. The security plan is now in accordance with the positions of Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," and ANSI N18.7-1973, "Industrial Security for Nuclear Power Plants," and conforms to the requirements of 10 CFR Part 50, Section 34(c) and 10 CFR Part 73, Section 40. The plan was submitted but withheld from public disclosure as provided for in Section 2.790 of 10 CFR Part 2.

We conclude that the industrial security program provides reasonable assurance that the risks associated with potential acts of sabotage that would lead to a significant threat to the public health and safety are acceptably low.

14.0 INITIAL TESTS AND OPERATION

Preoperational testing, fuel loading, initial criticality and approach to full power operation will be performed by the station operating personnel under the direct control of PG&E. A PG&E resident startup engineer will direct the program and will have sign-off responsibility, with the concurrence of the plant superintendent, for approval of test procedures and for evaluation of the completed test results. The Nuclear Steam System Supplier (Westinghouse) will furnish technical advice during the startup program. During hot functional testing, Westinghouse will provide a reactor coolant pump specialist, a chemist, and a quality assurance specialist for internals inspection. During fuel loading, a physicist, a chemist, and a fuel handling specialist will be on hand. During startup tests to power, a nuclear test engineer, a chemist, a transient analyst, and a reactivity computer instrumentation specialist are scheduled to be onsite.

The applicant has listed the tests to be performed during the test program, with a statement of the objective of each test. We have reviewed the program and conclude that it conforms to the positions stated in Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors." We conclude that the program described by the applicant will provide an adequate basis to confirm the safe operation of the plant, and is acceptable.

15.0 ACCIDENT ANALYSES

15.1 General

The staff and the applicant have evaluated the offsite radiological consequences for a number of postulated design basis accidents. These accidents are the same as those analyzed for previously licensed PWR plants and include a loss-of-coolant accident (including leakage from ESF components outside of containment), fuel-handling accident, a hydrogen purge dose (post-LOCA) accident, rod ejection accident, and rupture of a radioactive gas storage tank in the gaseous radioactive waste treatment system. All accidents have been evaluated at a core power level or 3423 MWt, with the exception of such accidents as the LOCA and those analyses for which adequacy of the containment and engineered safety features must be demonstrated. These accidents were evaluated at a core power level of 3580 MWt.

The offsite doses calculated by the staff for these accidents are presented in Table 15.1 of this report, and the assumptions used in these calculations are listed in Section 15.2. All potential offsite doses calculated by the applicant and the staff for the postulated accidents are within the guideline values of 10 CFR Part 100.

As part of the loss-of-coolant accident (LOCA), the staff and the applicant have also evaluated the consequences of leakage of containment sump water containing radioactive fission products which is circulated by the RHR system outside the containment after a postulated LOCA. We have assumed

the sump water to contain a mixture of iodine fission products in agreement with Regulatory Guide 1.7. About 0.4 hours after a postulated LOCA this water is pumped into the auxiliary building to be cooled by the residual heat removal heat exchanger. If a source of leakage should develop, such as a failure of the RHR pump seal, some water would leak into the auxiliary building. A portion of the iodine would become gaseous and would exit to the outside atmosphere after passing through the charcoal filters in the auxiliary building. The offsite doses resulting from such a sequence of events depends upon the temperature and magnitude of the assumed leakage. If the leakage occurred when the water temperature was below 212°F, a leak rate of less than 10 gpm over a period of one-half hour would result in doses (without filters) which could exceed the guideline values of 10 CFR Part 100. If the leakage occurred when the fluid is near its peak temperature of 248°F, then part of the leaking water would flash to steam, leading to additional iodine release. In this case, less than 1 gpm leakage for one-half hour would result in doses (without filters) which could exceed Part 100 guidelines.

If the charcoal filters in the auxiliary building are assumed to be effective in removing iodine, the offsite doses would be within the guidelines of 10 CFR Part 100, even for substantial amounts of leakage. As a result of the analyses discussed above, we will require that the auxiliary building filters conform to the requirements of engineered safety features (ESF) to the extent that electric heaters for humidity control will be required, and to the extent that the failure of any

single active component can be tolerated. The staff has agreed that elements of the filter system, such as the charcoal beds, need not be redundant. The applicant has been informed of our concern regarding potential RHR leakage, and is expected to propose appropriate design modifications. Resolution of this item will be reported on in a supplement to this Safety Evaluation Report.

On the basis of our experience with the evaluation of the steam line break and the steam generator tube rupture accidents for PWR plants of similar design, we have concluded that the consequences of these accidents can be controlled by limiting the permissible primary and secondary coolant system radioactivity concentrations so that potential offsite doses are small. We will include appropriate limits in the technical specifications on primary and secondary coolant activity concentrations. Similarly, we will include appropriate limits in the technical specifications on gas decay tank activity so that a single failure (such as sticking and lifting of a relief valve) does not result in doses that are more than a small fraction of the 10 CFR Part 100 guideline values.

15.2 Design Basis Accident Assumptions

15.2.1 Loss-of-Coolant Accident (Containment Leakage)

The assumptions used by the Regulatory staff in calculations of offsite doses from a LOCA were:

1. Power level of 3580 MWt.

2. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 1, June 1973.
3. Design containment leak rate of 0.10% per day for the first 24 hours and 0.05% per day thereafter.
4. Iodine removal by the containment quench spray system utilized the following parameters:

Primary Containment Volume	2.67×10^6 ft^3
Spray Fall Height	128 feet
Spray Flow Rate	2600 gpm
Elemental Mass Transfer Velocity	4.70 cm/sec
Organic Mass Transfer Velocity	0.0 cm/sec
Spray Drop Diameter	1500 microns
Spray Terminal Velocity	480 cm/sec
Fraction of Primary Containment	17%
Spray Reduction Limits:	
Elemental	[illegible]
Organic	1.0
Particulate	1000
Spray Removal Rates (for sprayed region only)	
Elemental	10.0 hrs^{-1}
Organic	0.0 hr^{-1}
Particulate	0.45 hr^{-1}

5. Ground level release with Pasquill Type "F" conditions and wind speed of 1.0 meter per second for short-term releases.

Our evaluation of the iodine removal effectiveness of the containment sprays is discussed further in Section 6.2.3 of this report.

15.2.2 Fuel Handling Accident

The assumptions used by the Regulatory staff to calculate offsite doses from a fuel handling accident were:

1. Rupture of all fuel rods in one assembly.
2. All gap activity the rods, assumed to be 10% of the noble gases and 10% of the iodine (with a peaking factor of 1.65), is released.
3. The accident occurs 100 hours after shutdown.
4. 99% of the iodine is retained in the pool water.
5. Iodine above the pool is 75% inorganic and 25% organic species.
6. Standard ground release meteorology and dose conversion factors.
7. Iodine removal factors of 90 and 70% for the charcoal filters for elemental and organic iodines, respectively.

15.2.3 Gas Decay Tank Rupture

The assumptions used by the Regulatory staff to calculate the offsite doses from a gas decay tank rupture were:

1. Gas decay tank contains one complete primary coolant loop inventory of noble gases resulting from operation with 1% failed fuel (75,000 curies of noble gases).
2. Standard ground level release meteorology and dose conversion factors.

15.2.4 Control Rod Ejection Accident

The assumptions used by the Regulatory staff to calculate offsite doses from a control rod ejection accident were:

Case I

1. Power Level of 3580 Mwt.
2. 10% fuel failed in transient.
3. 10% of iodine and noble gas inventory in gap of failed fuel.
4. Release of total gap activity in failed fuel to containment building.
5. 50% plate-out of radioactive iodines.
6. Containment building sprays are not initiated.
7. Containment building leak rate of 0.10% per day for 24 hours and one-half of this value thereafter.
8. Standard ground level release meteorology and dose conversion factors.

Case II

1. Power level of 3580 Mwt.
2. 10% fuel failed in transient.
3. 10% of iodine and noble gas activity in gap of failed fuel.
4. Release of total gap activity in failed fuel to primary coolant.
5. Primary to secondary coolant operational leakage is 1.0 gpm.
6. Loss of offsite power so that steam is released from secondary side relief valve.
7. Primary-secondary coolant equilibrium reached at 30 minutes after the accident.
8. Standard steam line release meteorology and dose conversion factors.

15.2.5 Hydrogen Purge Dose

The assumptions used by the Regulatory staff to calculate the exclusion boundary doses due to post-loss-of-coolant accident hydrogen purging were:

1. Power level of 3580 Mwt.
2. Containment Volume of 2.67×10^6 ft^3.
3. Purge Rate: 300 cfm (duration varies from 2 hrs per day at 28 days to 1 hr per day at 100 days).
4. Holdup Time Prior to Purging: 28 days.
5. Sodium Hydroxide Spray Reduction Factor for Iodine: 20.2.
6. Charcoal filter efficiencies of 90 and 70% for elemental and organic iodine, respectively.
7. X/Q Value: 4-30 days (1.75×10^{-5} sec/m^3).

TABLE 15.1

Potential Offsite Doses Due To Design Basis Accidents

Accident	Two Hour Exclusion Boundary (800 Meters)		Course of Accident Low Population Zone (9600 Meters)	
	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
Loss-of-Coolant* (Containment leakage)	151.	7.5	21.	<1
Post-LOCA Hydrogen Purge Dose	21.4	0.96		
Fuel Handling	23.	2.6	<1	<1
Gas Decay Tank Rupture	Negligible	2.6	Negligible	<1
Rod Ejection**				
Case I	19	<1	3.8	<1
Case II	19	1	3	<1

*The iodine fractions assumed to be released in the postulated loss-of-coolant accident are: elemental, 91%; organic, 4%; and particulate, 5%.

**See Section 15.2.4 of this report for the different assumptions used for Cases I and II.

16.0 TECHNICAL SPECIFICATIONS

The technical specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. We reviewed the proposed technical specifications in detail and have held a number of meetings with the applicant to discuss their contents. Modifications to the proposed technical specifications submitted by the applicant were made to describe more clearly the allowed conditions for plant operation. The finally approved technical specifications will be made part of the operating licenses. Included will be sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we have concluded that normal plant operation within the limits of the technical specifications will not result in potential offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

17.1 General

The description of the Quality Assurance (QA) Program for the operations phase of the Diablo Canyon Nuclear Power Plant, Units 1 and 2, is contained in Section 17.2 of the FSAR, as amended. Our evaluation of the QA Program for the operations phase is based on a review of this description, and on detailed discussions conducted with the applicant to determine if Pacific Gas and Electric Company's (PG&E's) QA Program for Diablo Canyon complies with the requirements of Appendix B of 10 CFR Part 50.

Our review of the applicant's QA Program included:

(1) A detailed evaluation of the QA Program contained in Section 17.2 of the FSAR, as amended.

(2) A meeting and discussions with PG&E representatives which resulted in revisions to the program description; these revisions were submitted in Amendments 4 thru 10 of the FSAR.

(3) Discussions with the Directorate of Regulatory Operations, Region V, relating to their review of PG&E's QA activities, and the implementation of a QA Program which complies with Appendix B of 10 CFR Part 50.

17.2 Organization

The PG&E organization for nuclear power generation, as it relates to QA activities, is shown in Figure 17.2-1 of the FSAR. This figure

BLANK PAGE

presents the company-wide organization and the Nuclear Plant Review and Audit Committee.

The President and chief operating officer has the responsibility for the overall management of the plant. The Senior Vice President (Engineering and Construction) provides technical support to the Electric Operations Department for plant modifications and other related activities, and is responsible for the development of the Quality Assurance Program. The Quality Assurance Program is directed and managed by the Director of the Quality Assurance Department.

A group of three committees is also assigned quality assurance responsibilities at the managerial level. These are: (1) the Plant Staff Review Committee (PSRC); (2) the General Office Nuclear Plant Review and Audit Committee (GONPR & AC); and (3) the President's Nuclear Advisory Committee (PNAC). These are identified in Figure 17.2-1 of the FSAR.

The PSRC is made up of the Plant Superintendent and his staff. The functions of this Committee are to review any change or test proposed for the plant, to review plant operating experience and to recommend disposition of major non-conformances. Minutes of the meetings are fowarded to the GONPR & AC.

The GONPR & AC has representatives from the Engineering, Steam Generation, Station Construction, and Quality Assurance Departments. The Committee reviews proposed changes in the facility or license, as well as PSRC reports of violations of procedures and

deviations. The GONPR & AC reguarly reviews the auditing activities of others and audits all plant activities. The Committee reports to management and the PNAC.

The PNAC members are the Manager of Claims and Safety (Chairman), the Senior Vice President of Engineering and Construction, the Assistant General Counsel, and the Quality Assurance Manager. The Committee will examine and report the activities of the GONPR & AC, review other audit activities and make independent investigations related to nuclear safety.

The Quality Assurance Department is organizationally independent of power plant operation. In the original submittal of the Diablo Canyon FSAR, the involvement of the QA Department in plant operations was limited to auditing. The Plant Superintendent was assigned responsibilities for conducting the quality control activities, whereby he would assign appropriate members of the plant staff to conduct the inspections. We questioned the applicant as to how adequate confidence of control of quality was obtained when the quality assurance functions are assigned to the same organization which executes the work. The applicant responded to our concern by changing these responsibilities to the Quality Assurance Engineer, and by providing the Quality Assurance Engineer with an inspection staff to be supervised at the plant. Furthermore, the applicant has stated that the QA Department has the authority to stop work, except in instances where stopping the work would

involve changing power level or separating a generating unit from the comapny's system grid. PG&E's reply also included the assignment of responsibilities to the QA Engineer and inspection staff to inspect modifications, repairs, replacement items, and reworked items. The inspectors on the site QA staff will have no other collateral duties and will be independent of personnel performing the work being inspected. The minimum qualifications and experience for the position of Quality Assurance Director have been described in the FSAR.

Based on our review of the QA organization of PG&E, as it affects the Diablo Canyon operations, we find adequate responsibility assigned to both onsite and offsite QA personnel and organizations. In connection with this responsibility, adequate authority has been assigned to the QA organization and to the Plant Superintendent to assure implementation of an effective QA Program. We conclude that sufficient organizational freedom exists for the QA engineer and inspection staff and for the Director of the QA Department to assure organizational independence and the objective implementation of QA activities.

17.3 Quality Assurance Program

The applicant's original FSAR did not adequately describe the QA Program for Operations. The description was limited in the areas of training, nonconformance control, inspection, welding stamp control, commitment to Regulatory guides, procurement document control, change control, and instrument calibration. Following discussion with the staff, PG&E amended their FSAR to provide an adequate description of the QA Program for Operations.

The QA Program for Operations is documented in the Quality Assurance Manual applicable to plant operations. The Manual contains written policies and procedures which cover all phases of plant operations, including operation maintenance, repair, modification testing, refueling, and procurement. The QA Manual is prepared, revised and distributed by the QA Department. In addition, the Plant Superintendent prepares, revises, and controls plant procedures which implement the Quality Assurance Manual applicable to plant operations. A list of the structures, systems, and components controlled by the QA Program has been identified in the FSAR.

A particular concern of the staff was the assignment of responsibility for the design and development of modifications of safety related systems, structures, and components (with the exception of major design) to the Manager of Steam Generation. PG&E responded to this concern with the establishment of a procedure which provides a practical and standardized means for making the determination, based on complexity of design together with reviews and checks required by the QA Program. The applicant has described a design control system to be utilized by the Manager of the Steam Generation Department to apply control measures for compatibility of materials, accessibility for inservice inspection, repair, and delineation of acceptance criteria. These measures will provide for review of applicable Regulatory standards, design bases, and licensing requirements.

Our concern for the lack of an independent qualified plant inspection staff was resolved in Amendment 8 to the FSAR. A staff of inspectors, supervised by the Quality Assurance Engineer, will now inspect, to instructions specified in an inspection plan, material and equipment received at the power plant, as well as the modifications, repairs, replacement items, and reworked items.

The applicant has described a procurement document control system supported by written procedures in the Quality Assurance Manual. The description of these procedures includes necessary measures to assure that the applicable Regulatory requirements, design bases, and reviews have been included in the procurement documents, except that these measures initially did not include a review by QA personnel. The FSAR has been amended to provide review of procurement documents for quality assurance provisions by the onsite QA Engineer. A QA record system has been described in the FSAR which requires that records of all significant activities be properly stored, retrieved, and retained at the plant.

PG&E has identified the implementation of a comprehensive system of planned and periodic audit functions. The QA Department, under the authority of the QA Director, has the assigned responsibility to verify compliance with the QA Program. Audits of plant operations, vendors, and suppliers are conducted in accordance with written procedures. Provisions have been made to utilize the assistance of personnel from

other departments, including persons from the Operating Department, provided they do not have direct involvement with the work being audited. Audit reports are submitted to the Senior Vice President (Engineering and Construction), the General Office Nuclear Plant Review and Audit Committee, the Manager of the Steam Generation Department, and the Plant Superintendent. Management audits are conducted by the General Office Nuclear Plant Review and Audit Committee. The President's Nuclear Advisory Committee will also make investigations at the request of corporate management.

A significant concern to the staff initially was the lack of commitment by the applicant to the applicable Regulatory Guides. However, in Amendment 8 to the FSAR, PG&E stated that they had conformed to the guidance provided in the document, "Guidance on Quality Assurance Requirements During Operations Phase of Nuclear Power Plant," (Orange Book) dated October 26, 1973, and the Regulatory Guides and Standards referenced therein. We find this to be acceptable.

In a Commission Memorandum and Order dated December 7, 1973, concerning the LaSalle County Nuclear Station, Units 1 and 2, the Regulatory staff was directed to determine, for facilities under construction and for construction applications under review, if quality assurance personnel have sufficient authority and organizational freedom to perform their critical functions effectively and without reservation.

Based on our evaluation of the information documented by PG&E in their letter dated February 6, 1974, and in Amendment 10 to the FSAR, we conclude that sufficient authority and organizational freedom exist within PG&E and Westinghouse to enable QA personnel to perform their critical functions effectively and without reservation for the design and construction of Diablo Canyon Units 1 and 2.

Based on our review of the QA Program controls which are being imposed on the operation of Units 1 and 2, and on our review of how these controls are being implemented, we conclude that the QA Program, as described in the FSAR, complies with the requirements of Appendix B of 10 CFR Part 50 and is acceptable for the operations phase of Units 1 and 2, including operation, maintenance, modification, repair, and refueling.

17.4 Conclusion

As a result of our detailed review and evaluation of PG&E's QA Program description contained in Section 17.2 of the FSAR and a series of discussions and meetings with the applicant, we conclude that the QA organization of PG&E has sufficient independence and authority to effectively conduct the QA Program without undue influence from those organizational elements responsible for cost and schedules. In addition, the QA Program description contains adequate QA provisions, requirements and controls demonstrating compliance with Appendix B of 10 CFR Part 50, and is, therefore, acceptable for controlling the operational phase of Units 1 and 2 of the Diablo Canyon Nuclear Plant.

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The report of the ACRS on the review of the application for operating licenses for Diablo Canyon Units 1 and 2 will be placed in the Commission's Public Document Room, and will be published by the Regulatory staff in a supplement to this Safety Evaluation Report. The supplement will be published prior to the final determination regarding issuance of operating licenses for the two units.

19.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are United States citizens. The applicant is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the requirements of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material required for military purposes is involved. For those reasons, and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish financial qualifications of an applicant for operating licenses are Section 50.33(f) and Appendix C of 10 CFR Part 50. We have reviewed the financial information presented in the application and the associated amendments regarding financial qualifications. Based on this review and consideration of financial data generally available to the financial analyst, we have concluded that Pacific Gas and Electric Company possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) to operate Units 1 and 2 of the Diablo Canyon Nuclear Power Plant, and if necessary, shut down the facilities and maintain them in a safe shutdown condition.

Funds to cover the estimated cost of operating the facilities are expected to be derived from sales of electric energy at rates which will cover all costs of production plus a reasonable return on invested capital.

The applicant's estimates of the annual cost of operating Units 1 and 2 are presented below. Unit costs (mills per kWH) are based on the following capacity factors for Unit 1: 1975 (commercial operation assumed to begin 9/1/75) - 32%; 1976 - 65%; 1977 - 75%; 1978 - 81%; 1979 - 83%; and 1980 - 83%. Capacity factors estimated for Unit 2 are as follows: 1976 (commercial operation assumed to begin 6/1/76 - 47%; 1977 - 66%; 1978 - 79%; 1979 - 83%; and 1980 - 83%. Maximum net electrical outputs assumed for Units 1 and 2 are 1131 and 1156 MWe,

respectively, and according to the applicant, are the expected outputs corresponding to the ultimate reactor outputs of 3488 and 3568 MWt, respectively, as discussed in Section 1.1 of the FSAR.

	Unit 1		Unit 2	
	Total Cost (millions)	Mills per kWH	Total Cost (millions)	Mills per kWH
1975	$ 0.9	0.3		
1976	85.2	13.2	$26.1	5.5
1977	88.0	11.8	66.9	10.1
1978	90.7	11.3	69.6	8.7
1979	93.7	11.4	72.7	8.6
1980	95.8	11.6	75.4	9.0

Averaging the estimated operating costs for Unit 1 during the years 1976-80 and for Unit 2 during the years 1977-80, when each unit will be in operation during the whole calendar year, produces the following amounts grouped by cost elements (A. + G. = administrative and general; O. + M. = operation and maintenance).

	Average Annual Cost (millions)		Mills per kWH	
	Unit 1 1976-80	Unit 2 1977-80	Unit 1	Unit 2
Nuclear fuel expense	$23.4	$24.0	3.0	3.0
Other nuclear power generation expense	2.3	1.2	0.3	0.2
Transmission expenses	0.2	0.1	---	---
A. + G. expenses	2.2	1.3	0.3	0.2
Total O. + M expenses	28.1	26.6	3.6	3.4
Depreciation	13.8	10.2	1.8	1.3
Taxes other than income taxes	5.1	3.7	0.7	0.5
Income taxes - Federal	7.3	4.0	1.0	0.5
Income taxes - other	1.8	1.1	0.2	0.1
Return @ 9.50%	34.6	25.5	4.5	3.2
Total cost	$90.7	$71.1	11.8	9.0

The 9.50% rate of return is the applicant's expected weighted cost of capital for new investments. System-wide sales of electric energy are expected to cover the costs shown above, which are equivalent to the "annual revenue requirements" of the subject facilities. The average annual costs of 11.8 and 9.0 mills per kWH for Units 1 and 2, respectively, as shown above, are below the applicant's average revenue of 18.3 mills per kWH received during 1973.

The applicant has estimated the total cost of decommissioning the subject facilities at $4.5 million for both units on the basis of 1974 dollars. Salvage value of useable equipment could partially offset the cost. The following activities are included in the cost estimate: (1) flushing and sealing of auxiliary systems outside the containment; (2) disposal of liquids and gases containing radioactive materials; (3) disposal of resins, filters, and miscellaneous radioactive materials; (4) post cleanup radiation surveys; and (5) sealing of the containment. Removal of spent fuel was considered part of the normal operating expenses and, therefore, was not included as a decommissioning cost. The applicant has indicated that essentially all of the above activities could be carried out by the normal operating crew of the two units in approximately one year. The applicant estimates that disposal of radioactive materials associated with decommissioning is equivalent to about three years of contracted waste disposal services during normal operation, and has included the disposal costs in the $4.5 million estimated

total cost of decommissioning. The applicant considers the decommissioning costs to be recoverable operating costs according to state regulatory policies, so that special reserves are not required.

The annual cost of maintaining the shutdown facilities in a safe condition is estimated at $160,000 on the basis of 1974 dollars, and provides for a permanent full time security force at the site plus periodic inspections and radiation surveys. The applicant considers the costs of maintaining the shutdown facilities in a safe condition to be recoverable operating costs according to state regulatory policies, so that special reserves are not required.

Information presented in Pacific Gas and Electric Company's annual report for 1973 indicates that operating revenues totaled $1490.2 million. Operating expenses were stated at $1172.6 million, of which $158.3 million represented depreciation. Interest on long-term debt was earned 2.8 times. Net income totaled $243.6 million, of which $149.7 million was distributed as dividends to stockholders, with the remaining $93.9 million retained for use in the business. As of December 31, 1973, the company's assets totaled $5471.1 million, most of which was invested in utility plant ($5109.9 million). Retained earnings amounted to $884.4 million. Financial ratios computed from the 1973 financial statements indicate an adequate financial condition, e.g., long-term debt to total capitalization - 51%, and to net utility plant - 50%; net plant to capitalization - 1.03; the operating ratio - 79%; and the rates of return

on common equity - 11.4%, on stockholders' investment - 10.0%, and on total investment - 7.1%.

The record of the company's operations during 1971-73 shows that operating revenues increased from $1260.3 million in 1971 to $1490.2 million in 1973; net income increased from $167.7 million to $243.6 million; net investment in utility plant increased from $4330.9 million to $5109.9 million; and the number of times interest on long-term debt was earned increased from 2.7 to 2.8. Moody's Investors Service rates the company's first and refunding mortgage bonds as Aa (high grade bonds). The company's Dun and Bradstreet rating is 5A1, the highest rating. Recent published data (Moody's Public Utility News Reports dated May 10, 1974) indicated that operating revenues increased from $1375.9 million for the 12 month period ended March 31, 1973 to $1532.1 million for the 12 month period ended March 31, 1974; net income increased from $220.3 million to $250.7 million; earnings available for common stock increased from $187.2 million to $213.0 million; and earnings per average common share outstanding increased from $3.07 to $3.27.

According to the "statement of aggregate net earnings" prepared in connection with the May 21, 1974 issuance of Series 74A, 9-1/8%, First and Refunding Mortgage Bonds, aggregate net earnings after income taxes for the 12 months ended April 30, 1974, amounted to $569,990,000 and were 3.40 times (versus the required 1.75) the $167,588,715 of annual interest charges on aggregate bonded indebtedness, including the

Series 74A bonds, as defined in the mortgage securing the bonds. As of December 31, 1973, the company had unreimbursed capital expenditures of $2.28 billion, of which $200 million was utilized as the basis for the issuance of the Series 74A bonds.

A summary analysis reflecting selected financial ratios and other pertinent data for Pacific Gas and Electric Company is included as Appendix B to this report.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licensees for facilities (such as power reactors) that are licensed under 10 CFR Part 50.

21.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The applicant, with respect to Diablo Canyon Units 1 and 2, is subject to the foregoing requirements, and will take the following steps, as required.

The applicant will furnish to the Commission proof of financial protection in the amount of $1,000,000 in the form of a nuclear energy liability insurance policy.

Further, the applicant will execute an Indemnity Agreement with the Commission as of the effective date of its pertinent preoperational fuel storage license. The applicant will pay the annual indemnity fee applicable to preoperational fuel storage.

Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which is $110 million.

Accordingly, no licenses authorizing operation of the Diablo Canyon Units 1 and 2 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement or amendment executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, in advance of anticipated issuance of the operating license documents, evidence in writing on behalf of the

applicant, that the present coverage has been appropriately amended and that the policy limits have been increased to an amount that meets the requirements of the Commission's regulations for reactor operation. Similarly, no operating licenses will be issued until an appropriate amendment to the present indemnity agreement has been issued. The applicant will be required to pay an annual fee for operating license indemnity as provided in the AEC regulations.

1.3 Conclusion

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140, concerning preoperational storage of fuel, are being satisfied and that, prior to issuance of any operating licenses, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and to execution of an appropriate indemnity agreement or amendment thereto with the Commission.

CONCLUSIONS

Based on our evaluation of the application as set forth above, it is our position that, upon favorable resolution of the outstanding matters described herein, we will be able to conclude that:

(1) The application for facility licenses filed by Pacific Gas and Electric Company dated October 2, 1973, as amended (Amendments Nos. 1 through 17), complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Part 1; and

(2) Construction of Units 1 and 2 (the facilities) has proceeded and there is reasonable assurance that it will be substantially completed, in conformity with Construction Permit Nos. CPPR-39 and CPPR-69, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(3) The facilities will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(4) There is reasonable assurance (a) that the activities authorized by the operating licenses can be conducted without endangering the health and safety of the public, and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Part 1; and

BLANK PAGE

(5) The applicant is technically and financially qualified to engage in the activities authorized by these licenses, in accordance with the regulations of the Commission set forth in 10 CFR Part 1; and

(6) The issuance of these licenses will not be inimical to the common defense and security or to the health and safety of the public.

Before operating licenses will be issued to Pacific Gas and Electric Company for operation of Units 1 and 2, the units must be completed in conformity with the provisional construction permits, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power levels must be verified by the Commission's Directorate of Regulatory Operations prior to issuance of the licenses.

Further, before operating licenses are issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY OF THE RADIOLOGICAL REVIEW

1. July 10, 1973	Application containing the FSAR tendered by Pacific Gas and Electric Company.
2. August 13, 1973	Applicant notified that the FSAR portion of the application is not sufficiently complete for docketing.
3. August 15, 1973	Initial site visit by LPM.
4. August 21, 1973	Meeting with applicant to discuss the deficiencies in the FSAR.
5. September 26, 1973	Revised application tendered by PG&E.
6. September 28, 1973	Applicant notified that application is sufficiently complete, and to file the appropriate documents as required by Section 50.30(c) of 10 CFR Part 50.
7. October 2, 1973	Application docketed.
8. October 10, 1973	Letter to applicant disclosing staff position regarding ATWS.
9. October 19, 1973	Notice of opportunity for hearing published in Federal Register (38 FR 29105).
10. October 25, 1973	Site visit and meeting related to geology and seismology.
11. November 5, 1973	Letter to applicant reminding him of his responsibility to maintain the local Public Document Room.
12. November 14, 1973	Site visit and meeting related to meteorology, hydrology, radiological assessment, and accident analysis.
13. November 19, 1973	Submittal of Amendment No. 1 consisting of miscellaneous revised and additional pages of the FSAR.

14. November 19, 1973	Staff notified by USGS of the discovery of possible offshore faults in the vicinity of Diablo Canyon.
15. December 11, 1973	Meeting with applicant to discuss electrical and instrumentation and control systems.
16. December 13, 1973	Submittal of preliminary geological information related to slope stability.
17. December 18, 1973	Meeting with potential intervenors in San Luis Obispo, California.
18. December 21, 1973	Letter to applicant confirming the safety review schedule for Diablo Canyon.
19. December 28, 1973	Request No. 1 to applicant for additional information concerning the site and certain radiological aspects of the plant.
20. January 4, 1974	Request No. 2 to applicant for additional information concerning the site and certain radiological aspects of the plant.
21. January 7, 1974	Response received from applicant to letter request of October 10, 1973 regarding ATWS.
22. January 8, 1974	Meeting with applicant and USGS in Menlo Park regarding offshore faults.
23. January 16, 1974	Request No. 3 to applicant for additional information concerning compliance with the Codes and Standards Rule, Section 50.55a of 10 CFR Part 50.
24. January 17, 1974	Submittal of Amendment No. 2 consisting of revised and additional pages of the FSAR, and providing responses to several items contained in the staff's acceptance review letter of August 13, 1973.
25. January 21, 1974	Site visit and meeting related to ECCS.
26. January 22, 1974	Letter informing applicant of the Commission's Memorandum and Order dated December 7, 1973, concerning the LaSalle County Nuclear Station, Units 1 and 2, and requesting information regarding the Quality Assurance Program.

27. January 25, 1974	ASLB Order indicating that an Operating License Hearing will be held for Diablo Canyon Units 1 and 2.
28. February 7, 1974	Request No. 4 to applicant for additional information concerning Radioactive Materials Safety.
29. February 12, 1974	Response received from applicant to letter request of January 22, 1974 regarding authority and organizational freedom in the Quality Assurance Program.
30. February 19, 1974	Submittal of Amendment No. 3 consisting of partial response to the staff's requests for additional information dated December 28, 1973 and January 4, 1974.
31. February 20-22, 1974	Site visit and meeting related to electrical and instrumentation and control systems.
32. February 22, 1974	Site visit and meeting related to geology and slope stability.
33. March 4, 1974	Submittal of Amendment No. 4 consisting of partial response to the staff's requests for additional information dated December 28, 1973, January 4, 1974 and January 16, 1974.
34. March 15, 1974	Letter to applicant requesting notification as to when information on outstanding items will be submitted.
35. March 18, 1974	Submittal of the Diablo Canyon Industrial Security Plan.
36. March 19, 1974	Submittal of Amendment No. 5 updating Chapters 4 and 15 of the FSAR concerning the 17x17 fuel design.
37. March 20, 1974	Letter to applicant requesting information on Bergen-Patterson snubbers which may be installed on any safety related systems.
38. March 20, 1974	Letter to applicant confirming that the Industrial Security Plan will be withheld from public disclosure in conformance with Section 2.790(d) of 10 CFR Part 2.

39. March 25, 1974	Submittal of the Diablo Canyon Site Emergency Plan.
40. March 26-27, 1974	Beginning of the first OL Prehearing Conference.
41. March 29, 1974	Submittal of Amendment No. 6 consisting of miscellaneous revised and additional pages of the FSAR.
42. April 12, 1974	Letter to applicant requesting additional information on the Industrial Security Plan.
43. April 12, 1974	Request No. 5 to applicant summarizing previously requested information for which acceptable responses have not been received.
44. April 15, 1974	Submittal of Amendment No. 7 consisting of miscellaneous revised and additional pages of the FSAR.
45. April 15, 1974	Request No. 6 to applicant summarizing previously requested information for which acceptable responses have not been received.
46. April 16, 1974	Letter from applicant responding to the staff's letter of March 15, 1974 regarding outstanding items.
47. April 24, 1974	Meeting with applicant to discuss electrical and instrumentation and control systems.
48. April 25, 1974	Meeting with applicant to discuss the Quality Assurance Program, the Industrial Security Plan, and the Site Emergency Plan.
49. April 26, 1974	Letter from applicant indicating when the information requested in our letters of April 12 and April 15, 1974 will be provided.
50. April 26, 1974	Meeting with applicant to discuss offshore geology and seismology.
51. April 30-May 1, 1974	Conclusion of the first OL Prehearing Conference.
52. May 7, 1974	Letter to applicant requesting additional financial information.

53. May 13, 1974	Submittal of Amendment No. 8 consisting of partial response to the staff's requests for additional information dated April 12 and April 15, 1974.
54. May 17, 1974	Letter from applicant requesting extension of the completion dates shown in the construction permits for Units 1 and 2 (CPPR-39 and CPPR-69, respectively).
55. May 21, 1974	Site visit and meeting related to pipe break outside containment.
56. May 23, 1974	Submittal of additional information on the Industrial Security Plan that was requested on April 12, 1974.
57. May 24, 1974	Letter to applicant requesting additional information regarding the extension of the construction permits for Units 1 and 2.
58. May 28, 1974	Submittal of additional electrical and instrumentation drawings.
59. May 31, 1974	Submittal of Amendment No. 9 consisting of partial response to the staff's requests for additional information dated April 12 and April 15, 1974.
60. May 31, 1974	Letter to applicant confirming that the revised Security Plan which was received on May 23, 1974 will be withheld from public disclosure in conformance with Section 2.790(d) of 10 CFR Part 2.
61. June 4, 1974	Meeting with applicant to discuss offshore geology and seismology.
62. June 4, 1974	Submittal of Amendment No. 10 consisting of additional information requested by the staff in connection with its review of the 17x17 fuel design.
63. June 6, 1974	Meeting with applicant regarding tsunami wave calculations.

64. June 17, 1974	Letter from applicant providing additional justification for their request for extension of construction permits CPPR-39 and CPPR-69.
65. June 17, 1974	Letter from applicant providing the additional financial information that was requested on May 7, 1974.
66. June 17, 1974	Request No. 7 to applicant for additional information on the 17x17 fuel design.
67. June 18, 1974	Letter to applicant requesting additional information regarding the preoperational testing program for the emergency core cooling system.
68. June 20, 1974	Submittal of Appendix I to the Site Emergency Plan consisting of the San Luis Obispo County Sheriff's Department Interim Evacuation Plan.
69. June 25, 1974	Response from applicant to our letter of March 20, 1974 concerning the use of Bergen-Patterson snubbers.
70. June 26, 1974	Letter to applicant concerning the scheduling of forthcoming operator and senior operator cold examinations for Unit 1 of the Diablo Canyon Nuclear Plant.
71. June 27, 1974	Submittal of Amendment No. 11 consisting of partial response to the staff's requests for additional information dated January 4, April 12, and April 15, 1974. Of particular importance in this amendment are the initial responses to questions on offshore geology and seismology.
72. June 28, 1974	Letter to applicant extending the latest dates for completion of construction for Units 1 and 2.
73. July 1, 1974	Submittal of four appendices which supplement the applicant's final report on the potential effects of pipe break outside containment.
74. July 2, 1974	Submittal of Amendment No. 12 consisting primarily of the applicant's final report on the potential effects of pipe break outside containment.

75. July 5, 1974	Meeting with applicant to review the progress of new geological field investigations related to offshore faults in the vicinity of the Diablo Canyon site.
76. July 5, 1974	Submittal of Amendment No. 13 consisting primarily of responses to the staff's request for additional information dated June 17, 1974 regarding the 17x17 fuel design.
77. July 12, 1974	Letter to applicant requesting additional information on the Industrial Security Plan.
78. July 16, 1974	Response from applicant to our letter of June 18, 1974 regarding preoperational testing of the ECCS.
79. August 2, 1974	Submittal of Amendment No. 14 consisting of miscellaneous revised and additional pages of the FSAR.
80. August 5, 1974	Submittal of Amendment No. 15 consisting primarily of the applicant's evaluation of compliance with the final ECCS acceptance criteria.
81. August 13, 1974	Response from applicant to our letter of July 12, 1974 requesting additional changes in the Industrial Security Plan.
82. August 16, 1974	Submittal of Amendment No. 16 consisting primarily of a partial response to the staff's request for information on tsunami waves caused by near-shore generators (see letter to applicant dated January 4, 1974). Amendment 16 also provides additional information in response to our letter of June 18, 1974 regarding preoperational testing of the ECCS.
83. September 3, 1974	Submittal of Amendment No. 17 consisting of miscellaneous revised and additional pages of the FSAR.
84. September 12, 1974	ACRS Subcommittee meeting emphasizing geology and seismology and ECCS - Appendix K evaluations.
85. September 18, 1974	Initial meeting regarding the Westinghouse - Standard Technical Specifications.
86. September 19, 1974	Site visit emphasizing items still outstanding in the safety review.

87. October 3, 1974 — Response from applicant to our letter of October 10, 1973, regarding ATWS.

88. October 10, 1974 — Request No. 8 to applicant for additional information on the operator requalification program.

APPENDIX B

PACIFIC GAS AND ELECTRIC COMPANY
FINANCIAL ANALYSIS
DOCKET NOS. 50-275 AND 50-323

(dollars in millions)

	Calendar Year Ended December 31		
	1973	1972	1971
Long-term debt	$2,539.0	$2,390.0	$2,301.0
Utility plant (net)	5,109.9	4,715.6	4,330.9
Ratio - debt to fixed plant	.50	.51	.53
Utility plant (net)	5,109.9	4,715.6	4,330.9
Capitalization	4,965.4	4,564.9	4,287.5
Ratio of net plant to capitalization	1.03	1.03	1.01
Stockholders' equity	2,426.4	2,175.0	1,986.5
Total assets	5,471.1	4,993.1	4,630.8
Proprietary ratio	.44	.44	.43
Earnings available to common equity	206.9	184.2	167.7
Common equity	1,811.5	1,610.0	1,521.5
Rate of earnings on common equity	11.4%	11.4%	11.0%
Net income	243.6	215.3	167.7
Stockholders' equity	2,426.4	2,175.0	1,986.5
Rate of earnings on stockholders' equity	10.0%	9.9%	8.4%
Net income before interest	337.3	345.3	309.8
Liabilities and capital	5,471.1	4,993.1	4,630.8
Rate of earnings on total investment	7.1%	6.9%	6.7%
Net income before interest	387.3	345.3	309.8
Interest on long-term debt	138.0	127.6	114.9
No. of times long-term interest earned	2.8	2.7	2.7
Net income	243.6	215.3	167.7
Total revenues	1,559.9	1,415.0	1,303.2
Net income ratio	.16	.15	.13
Total utility operating expenses	1,172.6	1,069.7	993.4
Total utility operating revenues	1,490.2	1,350.6	1,260.3
Operating ratio	.79	.79	.79
Utility plant (gross)	6,748.4	6,233.8	5,734.3
Utility operating revenues	1,490.2	1,350.6	1,260.3
Ratio of plant investment to revenues	4.53	4.62	4.55

[illegible]alization:

	1973		1972	
	Amount	% of Total	Amount	% of Total
Long-term debt	$2,539.0	51.1%	$2,390.0	52.4%
Preferred stock	614.9	12.4	564.9	12.4
Common stock & surplus	1,811.5	36.5	1,610.0	35.2
Total	$4,965.4	100.0%	$4,564.9	100.0%

Moody's Bond Rating: Aa

Dun & Bradstreet Credit Rating: 5A[illegible]

APPENDIX C

BIBLIOGRAPHY

(Documents referenced in or used to prepare the Safety Evaluation Report for the Diablo Canyon Nuclear Power Station, Units 1 and 2)

General

1. Preliminary Safety Analysis Reports with Amendments for the Diablo Canyon Nuclear Power Station, Units 1 and 2 (Docket Nos. 50-275 and 50-323).

2. Final Safety Analysis Report with Amendments 1 through 17 for the Diablo Canyon Nuclear Power Station, Units 1 and 2 (Docket Nos. 50-275 and 50-323).

3. United States Atomic Energy Commission Rules and Regulations, 10 CFR:

    Part 1, Statement of Organization and General Information
    Part 2, Rules of Practice
    Part 9, Public Record
    Part 20, Standards for Protection Against Radiation
    Part 50, Licensing of Production and Utilization Facilities
    Part 55, Operators' Licenses
    Part 71, Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions
    Part 73, Physical Protection of Special Nuclear Material
    Part 100, Reactor Site Criteria
    Part 140, Financial Protection Requirements and Indemnity Agreements
    Part 170, Fees for Facilities and Materials Licenses Under the Atomic Energy Act of 1954, as Amended.

4. United States Atomic Energy Commission Regulatory Guides.

5. National Environmental Policy Act of 1969 (NEPA).

Meteorology

6. Harris, M.F., "Effects of Tropical Cyclones Upon Southern California." Master of Arts Thesis, Department of Geography, California State University, Northridge Ca., 1973.

BLANK PAGE

7. Holzworth, G.C, "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States AP-101". Environmental Protection Agency, Office of Air Programs, Research Triangle Park, North Carolina, 1972.

8. Huschke, R.E., "Glossary of Meteorology." American Meteorological Society, Boston, Mass., 1959.

9. Korshover, J., "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1970." NOAA Technical Memorandum ERL ARL-34, Silver Spring, Md., 1971.

10. Sagendorf, J., "A Program for Evaluating Atmospheric Dispersion from a Nuclear Power Station." NOAA Technical Memorandum ERL ARL-42, Idaho Falls, Idaho, 1974.

11. SELS Unit Staff, National Severe Storms Forecast Center, "Severe Local Storm Occurrences, 1955-1967." ESSA Technical Memorandum WBTM FCST 12, Office of Meteorological Operations, Silver Spring, Md., 1969.

12. Slade, D.H. (ed.), "Meteorology and Atomic Energy - 1968." TID-24190, National Technical Information Service, Springfield, Va., 1968.

13. Smith, M.E. (ed.), "Recommended Guide for the Prediction of the Dispersion of Airborne Effluents." The American Society of Mechanical Engineers, New York, N.Y., 1968.

14. Thom, H.C.S., "Tornado Probabilities." Monthly Weather Review, October-December 1963, pp. 730-737, 1963.

15. Thom, H.C.S, "New Distributions of Extreme Winds in the United States." Journal of the Structural Division, Proceedings of the American Society of Civil Engineers - July 1968, pp. 1787 -1801, 1968.

16. Turner, D.B., "Workbook of Atmospheric Dispersion Estimates." Public Health Service Publication No. 999-AP-26, Cincinnati, Ohio, 1970.

17. United States Atomic Energy Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors-Revision 1." USAEC, Directorate of Regulatory Standards, Washington, D.C., 197[illegible].

18. United States Atomic Energy Commission, Regulatory Guide 1.23, "Onsite Meteorological Programs." USAEC, Directorate of Regulatory Standards, Washington, D.C., 1972.

19. United States Atomic Energy Commission, Regulatory Guide 1.42, "Interim Licensing Policy on As Low As Practicable For Gaseous Radioiodine Releases From Light-Water-Cooled Nuclear Power Reactors." USAEC, Directorate of Regulatory Standards, Washington, D.C., 1973.

20. United States Department of Commerce, Environmental Data Service: "Local Climatological Data, Annual Summary with Comparative Data - Santa Maria, California." Published annually through 1972.

21. United States Naval Weather Service, "Worldwide Airfield Summaries, Volume VIII Part 1, United States of America (West Coast, Western Mountains and Great Basin)." Federal Clearinghouse for Scientific and Technical Information, Springfield, Va., 1969.

Structural Engineering

22. American Institute of Steel Construction, "Specification for Design, Fabrication & Erection of Structural Steel for Buildings," (Sixth Edition) 101 Park Avenue, New York, New York, 10017, 1969.

23. American Concrete Institute, "Building Code Requirements for Reinforced Concrete (ACI 318-1963)," P.O. Box 4754, Redford Station, Detroit, Michigan 48219, 1963.

24. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," Section III, and Addenda, United Engineering Center, 345 East 47th Street, New York, New York 10017.

25. "Wind Forces on Structures", Final Report of the Task Committee on Wind Forces of the Committee on Load and Stresses of the Structural Division, Transactions of the American Society of Civil Engineers, 345 East 47th Street, New York, N.Y., 10017, Paper No. 3269, Vol. 126, Part II, p. 1124-1198, 1961.

Missile Protection

26. Gwaltney, R.C., "Missile Generation and Protection in Light-Water-Cooled Power Reactor Plants," USAEC Report ORNL-NSIC-22, September 1968.

27. Williamson, R.A. and Alvy, R.R., "Impact Effect of Fragments Striking Structural Elements," Holmes and Narver (Revised Edition), 1973.

Seismic Instrumentation

28. Kapur, Kanwar K., "Seismic Instrumentation for Nuclear Power Plants," ANS Topical Meeting on Water-Reactor Safety, Salt Lake City, Utah, March 1973.

29. United States Atomic Energy Commission, Regulatory Guide 1.12, "Instrumentation for Earthquakes." USAEC, Directorate of Regulatory Standards, Washington, D. C., 1974.

Component Design

30. American Society of Mechanical Engineers, "Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis in Sections III and VIII, Division 2," United Engineering Center, New York, New York, 1969.

31. American Society of Mechanical Engineers, "Nuclear Power Piping," USA Standard B 31.7, United Engineering Center, New York, New York, 1969.

32. American Society of Mechanical Engineers, "Power Piping," USA Standard B 31.1.0, United Engineering Center, New York, New York, 1967.

33. American Society of Mechanical Engineers, "Steel Pipe Flanges and Flanged Fittings," USA Standard B 16.5, United Engineering Center, New York, New York, 1968.

34. American Society of Mechanical Engineers, "Wrought Steel and Wrought Iron Pipe," USA Standard B 36.10, United Engineering Center, New York, New York, 1959.

35. American Society of Mechanical Engineers, "Wrought Steel Buttwelding Fittings," USA Standard B 16.9, United Engineering Center, New York, New York, 1964.

36. "Manufacturers Standardization Society Pipe Hangers and Supports - Materials and Design," MSS Standard Practice SP-58, Arlington, Virginia, 1959.

37. Manufacturers Standardization Society, "Pressure-Temperature Ratings for Steel Butt-Welding End Valves," MSS Standard Practice SP-66, New York, 1959.

38. American Water Works Association, "Standard for Steel Tanks Standpipes, Reservoirs, and Elevated Tanks for Water Storage," AWWA D 100-1967, New York New York, 1967.

39. "Standards of Tubular Exchange Manufacturers Association," Fifth Edition, New York, New York, 1968.

40. American National Standards Institute, "Design Basis for Protection of Nuclear Power Plants Against the Effect of Postulated Pipe Rupture," ANSI N 176 Draft, April 5, 1974.

Materials Engineering

41. American National Standards Institute, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors," ANSI N45.4-1972, New York, New York, March 16, 1972.

42. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code," Sections II, III and XI, with Addenda through Summer 1973, United Engineering Center, New York, New York.

43. American Society of Mechanical Engineers, "Methods and Definitions for Mechanical Testing of Steel Products," ASME-SA-370-71b, ASME Boiler and Pressure Vessel Code, Section II, Part A, Ferrous, 1971 Edition, with Addenda through Summer, 1972, New York New York.

44. American Society for Testing and Materials, "Copper-Copper Sulphate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, Part 31, Philadelphia, Pennsylvania, July 1973.

45. American Society for Testing and Materials, "Notched Bar Impact Testing Of Metallic Materials," ASTM-E-23-72, Annual Book of ASTM Standards, Part 31, Philadelphia, Pennsylvania, July 1973.

46. American Society for Testing and Materials, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," ASTM-E-208-69, Annual Book of ASTM Standards, Part 31, Philadelphia, Pennsylvania, July 1973.

47. American Society for Testing and Materials, "Surveillance Tests on Structural Materials in Nuclear Reactors," ASTM-E-185-73, Annual Book of ASTM Standards, Part 31, Philadelphia, Pennsylvania, July 1973.

48. United States Atomic Energy Commission Rules and Regulations, 10 CFR Part 50, and Appendices A, G, H, and J, and Paragraph 50.55a.

49. United States Atomic Energy Commission, Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," USAEC, Directorate of Regulatory Standards, Washington, D. C., October 1971.

50. United States Atomic Energy Commission, Regulatory Guide 1.31, "Control of Stainless Steel Welding - Revision 1," USAEC, Directorate of Regulatory Standards, Washington, D. C., June 1973.

51. United States Atomic Energy Commission, Regulatory Guide 1.34, "Control of Electro-Slag Weld Properties," USAEC, Directorate of Regulatory Standards, Washington, D. C., December 28, 1972.

52. United States Atomic Energy Commission, Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," USAEC, Directorate of Regulatory Standards, Washington, D.C., February 23, 1973.

53. United States Atomic Energy Commission, Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," USAEC, Directorate of Regulatory Standards, Washington, D. C., May 1973.

54. United States Atomic Energy Commission, Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steels," USAEC, Directorate of Regulatory Standards, Washington, D. C., May 8, 1973.

55. United States Atomic Energy Commission, Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," USAEC, Directorate of Regulatory Standards, Washington, D. C., May 1973.

56. United States Atomic Engery Commission, Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," USAEC, Directorate of Regulatory Standards, Washington, D.C., May 1973.

57. United States Atomic Energy Commission, Regulatory Guide 1.51, "Inservice Inspection of ASME Class 2 and 3 Nuclear Power Plant Components, "USAEC, Directorate of Regulatory Standards, Washington, D. C. May 1973.

58. United States Atomic Energy Commission Guideline Document, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspections," USAEC, Washington, D. C., January 31, 1969.

Reactor

59. Glasstone, S. and Sesonski, A., "Nuclear Reactor Engineering," D. Van Nostrand Company, Inc., Princeton, New Jersey, 1963.

60. Westinghouse Electric Corporation, "Reference Core Report 17 x 17," WCAP-8185, Pittsburgh, Pennsylvania, September 1973.

61. Tong, L.S., "Boiling Heat Transfer and Two-Phase Flow," John Wiley and Sons, Inc., New York, 1967.

62. United States Atomic Energy Commission, "Safety Evaluation Report of the Zion Nuclear Power Station, Units 1 and 2 (Docket Nos. 50-295 and 50-304)," USAEC, Directorate of Licensing, Washington, D. C., October 6, 1972.

63. United States Atomic Energy Commission, "Safety Evaluation Report of the Trojan Nuclear Power Station (Docket No. 50-344)," USAEC, Directorate of Licensing, Washington, D. C., October 1974.

64. Westinghouse Electric Corporation, "Anticipated Transients Without Reactor Trip in Westinghouse Pressurized Water Reactors," WCAP-8096, Pittsburgh, Pennsylvania, April 1973.

65. Barry, R.F., et al, "The Panda Code," WCAP-7759, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, April 1967.

66. Westinghouse Electric Corporation, "Safety Analysis of the 17 x 17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident," WCAP-8278 (Proprietary), Pittsburgh, Pennsylvania, February 1974.

67. Westinghouse Electric Corporation, "Hydraulic Flow Test of the 17 x 17 Fuel Assembly," WCAP-8236 (Proprietary), Pittsburgh, Pennsylvania, December 1973.

68. United States Atomic Energy Commission, "Technical Report on Densification of Westinghouse PWR Fuel," USAEC, Regulatory Staff, Washington, D.C., May 14, 1974.

Reactor Coolant System

69. Westinghouse Electric Corporation, "Overpressure Protection for Westinghouse Pressurized Water Reactors," WCAP-7769, Revision 1, Pittsburgh, Pennsylvania, 1972.

Engineered Safety Features

70. Allen, A. O., "The Radiation Chemistry of Water and Aqueous Solutions," Van Nostrand Company, 1961.

71. American Nuclear Society, "Decay Energy Release Rates Following Shutdown of Uranium-Fuel Thermal Reactors (DRAFT)," ANS Standard ANS-5.1, Hinsdale, Illinois, October 1971.

72. Brodrick, J. R., Burchill, W. E., and Lowe, P.A., "1/5 Scale Intact Loop Post-LOCA Steam Relief Tests," CENPD-63, (Rev.1), Combustion Engineering, Inc., March 1973.

73. Coward, H. F., and Jones, G. W., "Limits of Flammability of Gases and Vapors," Bureau of Mine Bulletin 503, 1952.

74. DiNunno, J. J., Anderson, F. D., Baker, R. E., and Waterfield, R. L., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, USAEC, Washington, D. C., March 23, 1962.

75. "FLOOD/MOD001 - A Code to Determine the Core Reflood Rate for a PWR Plant with 2 Core Vessel OUTLET Legs and 4 Core Vessel Inlet Legs," Interim Report, Aerojet Nuclear Company, November 2, 1972.

76. Lowe, P. A., Brodrick, J. R., and Burchill, W. E., "Steam-Water Mixing Test Program, Task D: Formal Report for Task A: 1/5 Scale Broken Loop," CENPD-65 (Rev. 1), Combustion Engineering, Inc., March 1973.

77. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Vol 87, p. 134, February 1965.

78. Parsly, L. F., "Design Considerations of Reactor Containment Spray Systems, Part VI, the Heating of Spray Drops in Air-Steam Atmospheres," USAEC Report ORNL-TM-2412, Oak Ridge, Tennessee, January 1970.

79. Rittig, W. H., Jayne, G. A., Moore, K. V., Slater, C. E., and Uptmer, M. L., "RELAP 3 - A Computer Program for Reactor Blowdown Analysis," IN-1321, Idaho Nuclear Corporation, June 1970.

80. Richardson, L. C., Finnegan, L. J., Wagner, R. J., and Waage, J. M., "CONTEMPT - A Computer Program for Predicting the Containment Pressure Temperature Response to a Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

81. Schmitt, R. C. Bingham, G. E., and Norberg, J. A., "Simulated Design Basis Accident Tests of the Carolina Virginia Tube Reactor Containment," Final Report, IN-1403, Idaho Nuclear Corporation, December 1970.

82. Slaughterbeck, D. C., "Comparison of Analytical Techniques Used to Determine Distribution of Mass and Energy in the Liquid and Vapor Regions of a PWR Containment Following a Loss-of-Coolant Accident," Special Interim Report, Idaho Nuclear Corporation, September 1970.

83. Slaughterbeck, D. C., "Review of Heat Transfer Coefifients for Condensing Steam in a Containment Building Following a Loss-of-Coolant Accident," IN-1388, Idaho Nuclear Corporation, September 1970.

84. Tagami, T., "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," Prepared for the National Reactor Testing Station, February 28, 1966 (Unpublished work).

85. Uchida, H., Oyama, A., and Toga, Y., "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," Proceeding of the Third International Conference on the Peaceful Uses of Atomic Energy Held in Geneva August 31 - September 9, 1964, Volume 13, Session 3.9, New York: United Nations 1965, (A/Conf. 28/P/436) pp. 93-104, May 1964.

86. Wagner, R. J., and Wheat, L. L., "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear, August 1973.

Instrumentation, Controls and Electric Power Systems

87. Final Safety Analysis Report for the Donald C. Cook Nuclear Plant, Units 1 and 2 - Sections 6, 7 and 8 (Docket Nos. 50-315 and 50-316).

88. Final Safety Analysis Report for the Zion Nuclear Power Station, Units 1 and 2 - Sections 6, 7 and 8 (Docket Nos. 50-295 and 50-304).

89. Institute of Electrical and Electronic Engineers, "Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE Std. 279-1971, New York New York.

90. Institute of Electric and Electronic Engineers, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," IEEE Std. 308-1971, New York, New York.

91. Institute of Electric and Electronic Engineers, "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electronic Equipment for Nuclear Power Generating Stations," IEE Std. 323-1971, New York, New York.

92. Institute of Electric and Electronic Engineers, "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE Std. 338-1971, New York, New York.

93. Institute of Electric and Electronic Engineers, "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Station Protection System," IEEE Std. 344-1971, New York, New York.

94. United States Atomic Energy Commission, Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems," USAEC, Directorate of Regulatory Standards, Washington, D. C., March 1971.

95. United States Atomic Energy Commission, Regulatory Guide 1.9., "Selection of Diesel Generator Set Capacity for Standby Power Supplies," USAEC, Directorate of Regulatory Standards, Washington, D. C., March 1971.

96. United States Atomic Energy Commission, Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," USAEC, Directorate of Regulatory Standards, Washington, D. C., February 1972.

97. United States Atomic Energy Commission, Regulatory Guide 1.32, Use of IEEE Std. 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," USAEC, Directorate of Regulatory Standards, Washington, D. C., August 1972.

98. United States Atomic Energy Commission, Regulatory Guide 1.40 "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," USAEC, Directorate of Regulatory Standards, Washington, D. C., March 1973.

99. United States Atomic Energy Commission, Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments," USAEC, Directorate of Regulatory Standards, Washington, D. C., March 1973.

100. United States Atomic Energy Commission, Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," USAEC, Directorate of Regulatory Standards, Washington, D. C., May 1973.

101. United States Atomic Energy Commission, "Safety Evaluation Report of the Donald C. Cook Nuclear Power Station, Units 1 and 2 (Docket Nos. 50-315 and 50-316)," USAEC, Directorate of Licensing, Washington, D. C., September 10, 1973.

102. United States Atomic Energy Commission, "Safety Evaluation Report of the Zion Nuclear Power Station, Units 1 and 2 (Docket Nos. 50-295 and 50-304)," USAEC, Directorate of Licensing, Washington, D. C., October 6, 1972.

103. Westinghouse Electric Corporation, "Seismic Testing of Electrical and Control Equipment (High Seismic Plants)," WCAP-8021, Pittsburgh, Pennsylvania, December 1971.

104. Westinghouse Electric Corporation, "Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)," WCAP-7744, Pittsburgh, Pennsylvania, August 1971.

105. Westinghouse Electric Corporation, "Solid State Logic Protection System Design," WCAP-7672, Pittsburgh, Pennsylvania, June 1971.

106. Westinghouse Electric Corporation, "Engineered Safeguards Final Device or Actuator Testing," WCAP-7705, Pittsburgh, Pennsylvania, March 1973.

Conduct of Operations

107. American National Standards Institute, "Industrial Security for Nuclear Power Plants," ANSI N 18.17-1973 (ANS 3.3), New York, New York, 1973.

108. American National Standards Institute, "Standard for Administrative Controls for Nuclear Power Plants," ANSI N 18.[illegible]72 (ANS 3.2), New York, New York, 1972.

109. American National Standards Institute, "Standards for Selection and Training of Personnel for Nuclear Power Plants," ANSI N 18.1-1971, New York, New York, 1972.

110. United States Atomic Energy Commission, Regulatory Guide 1.8, "Personnel Selection and Training," USAEC, Directorate of Regulatory Standards, Washington, D. C., March 10, 1971.

111. United States Atomic Energy Commission, Regulatory Guide 1.17, "Protection of Nuclear Power Plants Against Industrial Sabotage," USAEC, Directorate of Regulatory Standards, Washington, D. C., June 1973.

112. United States Atomic Energy Commission, Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)", USAEC, Directorate of Regulatory Standards, Washington, D. C., November 3, 1972.

113. International Atomic Energy Agency, "Planning for the Handling of Radiation Accidents," IAEA Safety Series No. 32, Vienna, Austria, November 1969.

114. United States Atomic Energy Commission, "Guide to the Preparation of Emergency Plans for Production and Utilization Facilities," USAEC, Washington, D. C., December 1970.

115. United States Atomic Energy Commission, "Utility Staffing and Training for Nuclear Power," WASH-1130 (Revision 1), USAEC, Washington, D. C., June 1973.

Initial Tests and Operation

116. United States Atomic Energy Commission, "Guide for the Planning of Initial Startup Programs," USAEC, Washington, D. C., December 17, 1970.

117. United States Atomic Energy Commission, "Guide for the Planning of Pre-operational Testing Programs," USAEC, Washington, D. C., December 7, 1970.

Radiation Protection

118. American National Standards Institute, "Guide for Administrative Practices in Radiation Monitoring, " USAS N 13.2-1969, New York, New York, 1969.

119. American National Standards Institute, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities," ANSI N 13.1-1969, New York, New York, 1970.

120. Environmental Instrumentation Group, "Survey of Instrumentation for Environmental Monitoring," Volume 3 - Radiation, Lawrence Berkeley Laboratory, Berkeley, California, October 1973.

Waste Management

121. United States Atomic Energy Commission, "Final Environmental Statement Concerning Proposed Rule Making Action: Numerical Guides For Design Objectives and Limiting Conditions To Meet The Criterion "As Low As Practicable" For Radioactive Material In Light-Water-Cooled Nuclear Power Reactor Effluents," WASH-1258, USAEC, Washington, D. C., July 1973.

END

DATE FILMED

1 / 9 / 75