Southern California Edison Company

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December 15, 1995

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TELEPHONE
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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: **Docket Nos. 50-361 and 50-362**
Response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE)
San Onofre Nuclear Generating Station
Units 2 and 3

Reference: Letter from Walter C. Marsh, dated March 24, 1995, Response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities," San Onofre Nuclear Generating Station Units 2 and 3

This letter provides Southern California Edison's (Edison's) response to Generic Letter 88-20, Supplement 4, Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR 50.54(f), as committed to in the Reference. The San Onofre Nuclear Generating Station (SONGS) IPEEE program meets the objectives of Generic Letter 88-20, Supplement 4 and NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated June 1991.

As a result of the IPEEE effort, several procedural and design modifications were identified as cost beneficial in enhancing the ability of SONGS Units 2 and 3 to resist core damage due to external initiating events. Some of these modifications have already been completed. All modifications will be completed by the end of the next refueling outage for each unit. The Unit 2 Cycle 9 refueling outage is currently scheduled for November, 1996, and the Unit 3 Cycle 9 refueling outage is currently scheduled for March, 1997.

As documented in the enclosure, the core damage frequency due to external initiating events projected for Cycle 9 operation is approximately 3.3E-5 per year.

This submittal completes Edison's response to GL 88-20, Supplement 4 for the IPEEE. If you have any questions regarding this report, please call me.

Respectfully submitted,

Walter C. Marsh

State of California
County of San Diego
On Dec. 15, 1995 before me, Linda L. Rulon , personally appeared Walter C. Marsh, personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.

WITNESS my hand and official seal.

LINDA L. RULON
COMM. #958229
Notary Public-California
ORANGE COUNTY
My comm. expires MAR 04,1996

Signature Linda L. Rulon

Enclosure

cc: L. J. Callan, Regional Administrator, NRC Region IV
J. E. Dyer, Director, Division of Reactor Projects, Region IV
K. E. Perkins, Jr., Director, Walnut Creek Field Office, NRC Region IV
J. A. Sloan, NRC Senior Resident Inspector, San Onofre Units 2 & 3
M. B. Fields, NRC Project Manager, San Onofre Units 2 and 3

INDIVIDUAL PLANT EXAMINATION
OF EXTERNAL EVENTS
FOR
SAN ONOFRE NUCLEAR GENERATING STATION
UNITS 2 & 3

IN RESPONSE TO
GENERIC LETTER 88-20, SUPPLEMENT 4

Submittal Document

Southern California Edison

December 1995

TABLE OF CONTENTS

TABLE OF ACRONYMS

ADV	Atmospheric Dump Valve
AFW	Auxiliary Feedwater
AOI	Abnormal Operating Instruction
AOV	Air-Operated Valve
ASEP	Accident Sequence Evaluation Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BAMU	Boric Acid Makeup
CC	Common Cause
CCAS	Containment Cooling Actuation Signal
CCDP	Conditional Core Damage Probability
CCW	Component Cooling Water
CDF	Core Damage Frequency
CE	Combustion Engineering
CEDM	Control Element Drive Mechanism
CEFC	Containment Emergency Fan Cooler
CEOG	Combustion Engineering Owners Group
CET	Containment Event Tree
CUES	Central and Eastern United States
CIAS	Containment Isolation Actuation Signal
CIS	Containment Isolation System
CLASSI	Foundation Impedance Computer Program
COV	Lognormal Distribution Coefficient of Variation
CPIS	Containment Purge Isolation Signal
CRIS	Control Room Isolation Signal
CRS	Control Room Supervisor
CS	Containment Spray
CSAS	Containment Spray Actuation Signal
CSR	Containment Spray Recirculation
CST	Condensate Storage Tank
CVCS	Chemical Volume and Control System
CYLREC	Embedment Correction Terms Computer Program
DBE	Design Basis Earthquake
DGB	Diesel Generator Building
DPD	Discrete Probability Distribution
ECCS	Emergency Core Cooling System
ECW	Emergency Chilled Water
EDG	Emergency Diesel Generator
EET	Extended Event Tree
EFAS	Emergency Feedwater Actuation Signal

EHC	Electro-Hydraulic Control
EOI	Emergency Operating Instruction
EQE	EQE International
EQESRA	EQE Seismic Quantification Code
ERIN	ERIN Engineering and Research, Inc.
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FAI	Fauske and Associates, Inc.
FHIS	Fuel Handling Building Isolation Actuation Signal
FIVE	Fire Induced Vulnerability Evaluation
FMEA	Failure Modes and Effects Analysis
FRSS	Fire Risk Scoping Study
FTAP	Fault Tree Analysis Program
FWIS	Feedwater Isolation Signal
FWIV	Feedwater Isolation Valve
GERS	Generic Equipment Ruggedness Spectra
GL	Generic Letter
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HPSI	High Pressure Safety Injection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilating, and Air Conditioning
IEEE	Institute of Electrical and Electronics Engineers
INPO	Institute of Nuclear Power Operations
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISA	Instrument & Service Air
ISLOCA	Interfacing System Loss of Coolant Accident
LCO	Limiting Condition For Operation
LDC	Initiator/Event Tree Reference to Loss of 125V DC
LL	Initiator/Event Tree Reference to Large LOCA
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
LOP	Loss of Offsite Power
LOSP	Loss of Offsite Power
LOVS	Loss of Voltage Signal
LPSI	Low Pressure Safety Injection
MAAP	Modular Accident Analysis Program
MFW	Main Feedwater
M/G	Motor Generator
MGL	Multiple Greek Letter
ML	Initiator/Event Tree reference to Medium LOCA

MLLW	Mean Lower Low Water
MOV	Motor-Operated Valve
MSIS	Main Steam Isolation Signal
MSIV	Main Steam Isolation Valve
MSS	Main Steam System
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NEDO	Nuclear Engineering Design Organization (SCE)
NEP	Non-Exceedance Probability
NRC	Nuclear Regulatory Commission
NSG	Nuclear Safety Group (SCE)
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resource Council
OI	Operating Instruction
P&ID	Piping and Instrumentation Diagram
PCS	Power Conversion System
PDS	Plant Damage State
PG&E	Pacific Gas & Electric
PIV	Pressure Isolation Valve
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PORV	Power Operated Relief Valves
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
PSA	Peak Spectral Acceleration
PSF	HRA Performance Shaping Factors
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PZR	Pressurizer
QA	Quality Assurance
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
REBECA	Reliability Engineering Building-Block Environment for Computer Analysis
REI	Risk Engineering, Inc.
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTO	Reactor Trip Override
RWST	Refueling Water Storage Tank
SA	Spectral Acceleration
SBO	Station Blackout
SCE	Southern California Edison

SCOZD	Southern California Offshore Zone of Deformation
SDC	Shutdown Cooling
SDCHX	Shutdown Cooling Heat Exchangers
SDS	Seismic Damage State
SEQPRO	Sequence Processor
SET	Seismic Event Tree
SEL	Seismic Equipment List
SEWS	Screening and Evaluation Worksheets
SG	Steam Generator
SGR	Initiator/Event Tree reference to SGTR
SGTR	Steam Generator Tube Rupture
SHAKE	Soil Properties Computer Program
SI	Safety Injection
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SL	Small LOCA
SLB	Steam Line and Feedwater Line Breaks
SLOCA	Small LOCA
SNL	Sandia National Laboratories
SOG	Seismic Owner's Group
SONGS	San Onofre Nuclear Generating Station
SPTA	Standard Post-Trip Actions
SQUG	Seismic Qualification Utility Group
SRL	Seismic Relay List
SRP	Standard Review Plan
SSC	Safe Shutdown Component
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSL	Small-Small LOCA
SSMRP	Seismic Safety Margins Research Program
STA	Shift Technical Advisor
SV	Safety Valve
SWC	Saltwater Cooling
TDAFWP	Turbine Driven Auxiliary Feedwater Pump
TGIS	Toxic Gas Isolation System
TT	Transient With PCS Initially Available
TWS	Initiator/Event Tree reference to Anticipated Transient Without Scram
UFHA	Updated Fire Hazard Analysis
UFSAR	Updated Final Safety Analysis Report
UHS	Uniform Hazard Spectra
UPS	Uninteruptible Power Supply
USI	Unresolved Safety Issue

VCT	Volume Control Tank
VL	Initiator/Event Tree reference to ISLOCA
VR	Initiator/Event Tree reference to Reactor Vessel Rupture
WCC	Woodward-Clyde Consultants
fps	feet per second
gpm	gallons per minute
pcm	10^{-5} dp
ppm	parts per million
psia	pounds per square inch (atmosphere)
psig	pounds per square inch (gauge)

1. EXECUTIVE SUMMARY

This report documents the results and findings of the Individual Plant Examination of External Events (IPEEE) for San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 in response to Nuclear Regulatory Commission (NRC) Generic Letter 88-20, Supplement 4. This summary covers the 1) background and objectives, 2) plant familiarization, 3) overall methodology, and 4) summary of major findings.

1.1 BACKGROUND AND OBJECTIVES

In November of 1988, the NRC issued Generic Letter 88-20 [1-1] requesting that all U.S. nuclear utilities perform a plant-specific Individual Plant Examination (IPE) for severe accident vulnerabilities. This effort involved an integrated analysis of plant and system response to a wide spectrum of internal, randomly initiated events such as reactor trips, loss of offsite power, and loss of coolant accidents (LOCAs) with an emphasis on quantification of plant core damage frequency and evaluation of containment performance. Southern California Edison (SCE) completed and submitted the IPE to the NRC on April 29, 1993.

In June of 1991, the NRC issued Supplement 4 to Generic Letter 88-20 [1-2] requesting that all U.S. nuclear utilities perform a plant specific IPE of external events (IPEEE) to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee determined improvements and corrective actions to the NRC. The IPEEE initiators include earthquakes, internal fires, high winds and tornados, external floods, and transportation & nearby facility accidents. The specific objectives of the IPEEE study, which are similar to the IPE, were for each licensee:

(1) to develop an appreciation of severe accident behavior,

(2) to understand the most likely severe accident sequences that could occur at its plant under full power conditions,

(3) to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and

(4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

1.2 PLANT FAMILIARIZATION

SONGS is located on the coast of Southern California, in San Diego County, approximately 62 miles southeast of Los Angeles and approximately 51 miles northwest of San Diego. The station is located entirely within the Camp Pendleton Marine Corps Base. The station includes three reactors. Unit 1, which operated until late 1992, is located northwest of and immediately adjacent to Units 2 and 3. Units 2 and 3 are essentially identical operating units and have the provision for limited sharing of AC power systems and cooling water intake structures. The plants also share a common control room complex, radwaste facilities, instrument air/nitrogen and emergency heating, ventilation, and air conditioning (HVAC) systems. Other than the above noted commonalities SONGS 2 and 3 operate as independent entities.

SONGS Unit 2 received its operating license in February 1982 and began commercial operation in August 1983. SONGS Unit 3 received its operating license in November 1982 and began commercial operation in April 1984.

1.3 OVERALL METHODOLOGY

SCE completed the IPEEE in accordance with Generic Letter 88-20, Supplement 4 and NUREG-1407 [1-2,1-3] using methods consistent with NUREG/CR-2300 and NUREG/CR-4840 [1-4,1-5]. Separate methodologies were used to address each initiating event (i.e., seismic, internal fires, high winds, floods, and other hazards). The general methodology is described below with specifics provided in the methodology subsections of Sections 3, 4, and 5.

The seismic evaluation was completed in accordance with seismic probabilistic risk analysis (PRA) methods found in NUREG/CR-2300 [1-4]. A plant-specific seismic hazard curve was developed by a team of seismic consultants and reviewed by a panel of independent experts [1-6]. A team of in-house and consultant engineers combined to assess component, structural, and relay fragilities. Relay chatter impact was assessed for non-seismically screened relays. The combined information was logically assembled to produce the SONGS 2/3 seismic event tree which is the plant seismic model. Using the SONGS 2/3 seismic event tree model, the hazard curve was mathematically convolved with developed seismic fragility curves for safety equipment that are required to operate following a seismic event. A supplemental analysis of non-seismic failures and human actions was then integrated with the seismic analysis. The results of the seismic analysis include the seismic core damage frequency and an understanding of the relative seismic risk frequencies for various pathways to core damage. Core damage scenarios greater than 1E-7/year were evaluated for potential containment and containment systems vulnerabilities.

The internal fire analysis was completed in accordance with screening techniques found in the Nuclear Management and Resources Council/Electric Power Research Institute (NUMARC/ EPRI) Fire Induced Vulnerability Evaluation (FIVE) methodology [1-7] and PRA methods found in NUREG/CR-2300. The FIVE methodology's three phase screening process screens out compartments whose core damage risk is less than 1E-6/year. Using the expanded Level 2 containment analysis from the internal events IPE, the dominant scenarios from the last screening phase (phase III) were evaluated with the SONGS containment event tree and logic models to assess potential containment or containment systems vulnerabilities.

The high winds, floods, and transportation and nearby facility accidents hazards analysis employed a screening approach as outlined in NUREG-1407 to assess whether SONGS meets 1975 Standard Review Plan (SRP) criteria. If SONGS did not meet the 1975 SRPcriteria, further evaluation would be required to assess the risk significance of non-conforming items. As in the seismic and internal fires analysis, containment performance was assessed.

1.4 SUMMARY OF MAJOR FINDINGS

The subsequent sections of this summary present the results of the IPEEE.

1.4.1 LEVEL I RESULTS

The total mean core damage frequency (CDF) for *external* events at SONGS 2/3 was calculated to be approximately 3.3E-5/year[1]. (The CDF for *internal* event initiators [loss of coolant accidents (LOCAs), steam generator tube ruptures (SGTRs), loss of offsite power (LOP), etc.] was reported in the SONGS 2/3 IPE to be 3.0E-5/year.)

The level I results for seismic and internal fire hazards are listed in Table 1.4-1. Additional results information for the seismic and fire hazards risk analyses are provided in Sections 1.4.1.1 and 1.4.1.2.

---

[1] Following completion of all scheduled modifications discussed in Section 1.4.3.

TABLE 1.4-1
LEVEL I RESULTS

INITIATING EVENT	MEAN CORE DAMAGE FREQUENCY
Seismic	1.7E-5/year
Internal Fires[2]	1.6E-5/year

[2] This value represents the core damage frequency of all unscreened core damage scenarios.

Core damage frequency for high winds, floods, and other hazards was not quantified since the results of the high winds, floods, and other hazards events analysis demonstrated that SONGS 2/3 meets the 1975 Standard Review Plan criteria, and that the analyzed and screened events do not pose a significant risk. In accordance with NUREG-1407, this analysis demonstrates that "no other plant-unique external event is known that poses any significant threat of severe accident within the context of the screening approach for 'High Winds, Floods and Others'".

1.4.1.1 Seismic

The SONGS 2/3 mean seismic core damage frequency is 1.7E-5/year. The seismic risk is evaluated for the seismic hazard developed specifically for SONGS 2/3 which includes accelerations beyond the SONGS 2/3 design basis earthquake of 0.67g peak ground acceleration (PGA). The risk, as a function of average spectral acceleration between 1 and 10 Hz, is provided in Figure 1.4-1. The "average spectral acceleration" is approximately 2.3 times the "peak ground acceleration". Average spectral acceleration is used because it is necessary to anchor the hazard curve to the same parameter used for component fragilities. The dominant contributors by sequence are presented in Table 1.4-2.

FIGURE 1.4-1
SEISMIC EVENT MAGNITUDE VS. CDF

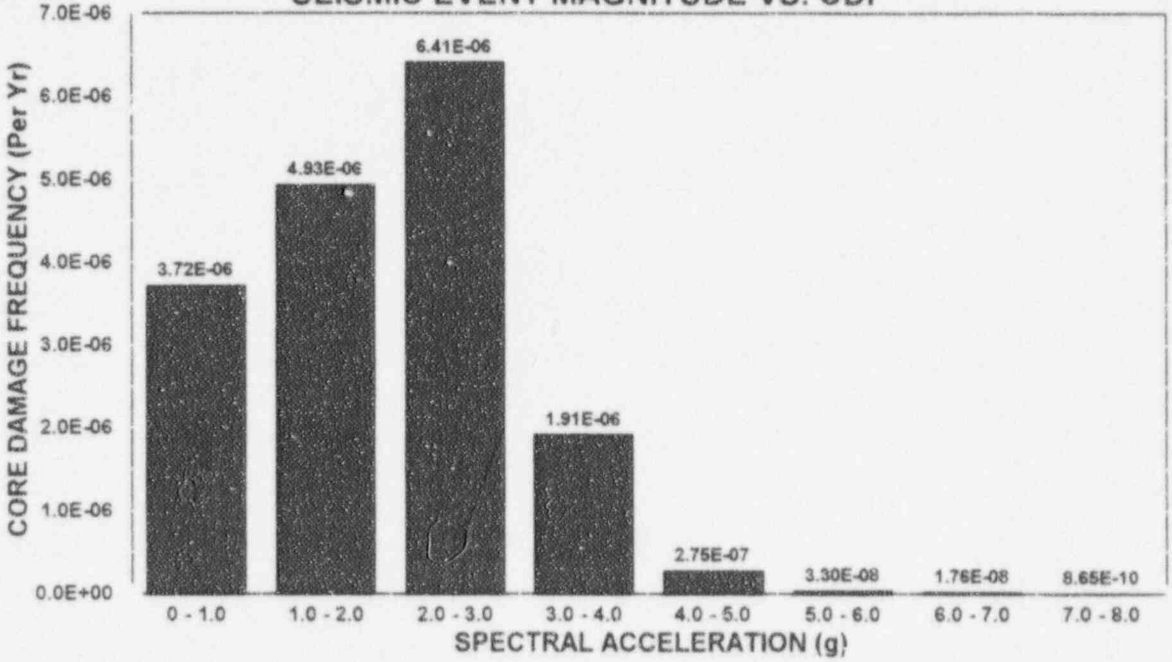

TABLE 1.4-2
SONGS 2/3 TOP SEISMIC CORE DAMAGE SEQUENCES

Seq. Rank #	SDS	Sequence Description	Seismic Failures	Dominant Random Failures	SDS Frequency	CCDP	Sequence CDF
1	20	Seismic Loss of Offsite Power Seismic Failure of Emergency Switchgear	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) 480V Switchgear Motor Control Centers	N/A	6.57E-6	1.0	6.57E-6
2	29	Seismic Loss of Offsite Power Seismic Failure of Instruments/Control	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Auxiliary Building Emergency Chillers	N/A	3.31E-6	1.0	3.31E-6
3	16	Seismic Loss of Offsite Power No Seismic Failure of Other Components Random Loss of Emergency Diesel Generators	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter)	Emergency Diesel Generators DG Emergency Supply Fans DG Fuel Transfer Pumps	6.94E-3	3.57E-4	2.48E-6
4	16	Seismic Loss of Offsite Power No Seismic Failure of Other Components Random Loss of AFW	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter)	Operator failure - Condensate M/U Turbine Driven AFW Pump Motor Driven AFW Pumps Operator Failure - Battery Chrgr Failure TD AFW Pump Control Valves Emergency Chillers	6.94E-3	1.89E-4	1.31E-6
5	26	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of Emergency Switchgear	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA 480V Switchgear Motor Control Centers	N/A	1.15E-6	1.0	1.15E-6
6	23	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of Recirculation	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA Emergency Sump Valve Bellows	N/A	8.01E-7	1.0	8.01E-7

TABLE 1.4-2
SONGS 2/3 TOP SEISMIC CORE DAMAGE SEQUENCES

Seq. Rank #	SDS	Sequence Description	Seismic Failures	Dominant Random Failures	SDS Frequency	CCDP	Sequence CDF
7	18	Seismic Loss of Offsite Power Seismic Failure of CCW/SWC	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Safety Equipment Building CCW Heat Exchangers SWC Valve Relays (Chatter) Primary Plant M/U Tank SWC Discharge Gate	N/A	5.37E-7	1.0	5.37E-7
8	19	Seismic Loss of Offsite Power Seismic Failure of Condensate Storage Tanks	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Condensate Storage Tank T120 Condensate Storage Tank T121	N/A	4.56E-7	1.0	4.56E-7
9	21	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Random Loss of CCW Random Loss of Safety Injection	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA	CCW Heat Exchangers CCW Non-Critical Loop Isolation Vlvs CCW Pumps High Pressure Safety Injection Pumps	1.49E-5	2.13E-2	3.18E-7
10	24	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of CCW/SWC	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA Safety Equipment Building CCW Heat Exchangers SWC Valve Relays (Chatter) Primary Plant M/U Tank SWC Discharge Gate	N/A	1.87E-7	1.0	1.87E-7

1.4.1.2 Internal Fires

Using the FIVE methodology's three phase screening process, ten fire compartments were identified as having a core damage frequency greater than the screening criteria of 1E-6/yr. Further, more rigorous calculations that utilized a cutset culling limit consistent with that of the IPE reduced the CDF for all ten compartments such that four of the ten compartments are less than 1E-6/year. The combined CDF for the ten compartments is 1.6E-5/year. Table 1.4-3 provides the risk contributions of these compartments.

TABLE 1.4-3
CDF OF FIRE COMPARTMENTS NOT SCREENED BY FIVE METHODOLOGY

FIRE COMPARTMENT	DESCRIPTION	CORE DAMAGE FREQUENCY
2-AC-50-40	Switchgear Room	3.3E-6
2-AC-50-35	Switchgear Room	2.9E-6
2-TB-148	Turbine Building	2.2E-6
2-PE-63-3B	Electrical Penetration	1.7E-6
3-PE-63-3B	Electrical Penetration	1.6E-6
2-PE-45-3A	Electrical Penetration	1.0E-6
2-AC-85-71	Switchgear Room	0.94E-6
2-DG-30-158	Diesel Generator Room	0.93E-6
2-DG-30-155	Diesel Generator Room	0.93E-6
2-AC-9-17	Relay Room	0.92E-6
Total CDF of Fire Compartments Not Screened By FIVE		1.6E-5/yr

1.4.2 LEVEL II RESULTS

Generic Letter 88-20, Supplement 4, Appendix 2 states:

> "The evaluation of the containment performance (Level II analysis) for external events should be directed toward a systematic examination of whether there are sequences that involve containment failure modes distinctly different from those identified in the IPE internal events evaluation or contribute significantly to the

likelihood of functional failure of the containment (seismic-induced loss of containment barrier independent of core melt)."

The following sections discuss the comparative findings.

1.4.2.1 Seismic

Results of the SONGS 2/3 seismic level II analysis are summarized in Table 1.4-4. Results show that seismic-induced containment bypass with greater than 0.1% volatiles released (categories D and T) is extremely unlikely. The small increase in frequency of release categories L, B, and G results from the dependence of containment heat removal systems on AC power combined with an appreciable likelihood of a loss of offsite power following a seismic event.

Based on comparison with the IPE results, there are no significant additional seismic-induced large early release source terms and no seismic-induced containment bypass sequences. Also, there are no additional seismic-induced containment bypass, isolation or other containment failure modes.

TABLE 1.4-4
RELEASE CATEGORY AND PROBABILITY OF SEISMIC IPEEE

RELEASE CATEGORY	RELEASE CATEGORY DEFINITION	SEISMIC IPEEE RELEASE FREQUENCY (PER YEAR)	IPE RELEASE FREQUENCY (PER YEAR)
S	Success, no containment failure within 48 hours, < 0.1% volatiles released	9.1E-6	2.6E-5
L	Late containment failure, up to 1% volatiles released	7.5E-6	2.2E-6
W	Late containment failure, more than 10% volatiles released	2.4E-8	6.9E-7
G	Early/isolation failure, containment failure prior to or at vessel failure, up to 10% volatiles released	3.9E-7	2.0E-8
B	Containment bypassed, < 0.1% volatiles released	2.6E-7	2.2E-7
D	Containment bypassed, up to 10% volatiles released	0	1.2E-6
T	Containment bypassed, > 10% volatiles released	0	6.5E-7

1.4.2.2 Fire

Results of the SONGS 2/3 internal fire level II analysis are summarized in Table 1.4-5. Results show that fire-induced containment bypass with greater than 0.1% volatiles released (categories D and T) is extremely unlikely. The frequency for release categories L, B, and W for internal fires and IPE are relatively similar. The increase in frequency of release category G is due to the model changes incorporated by the expanded Level 2 containment analysis.

Based on comparison with the IPE results, there are no significant additional fire-induced large early release source terms and no fire-induced containment bypass sequences. Also there are no additional fire-induced containment bypass, isolation or other containment failure modes.

TABLE 1.4-5
RELEASE CATEGORY AND PROBABILITY OF FIRE IPEEE

RELEASE CATEGORY	RELEASE CATEGORY DEFINITION	FIRE IPEEE RELEASE FREQUENCY (PER YEAR)	IPE RELEASE FREQUENCY (PER YEAR)
S	Success, no containment failure within 48 hours, < 0.1% volatiles released	1.4E-5	2.6E-5
L	Late containment failure, up to 1% volatiles released	2.4E-6	2.2E-6
W	Late containment failure, more than 10% volatiles released	2.0E-7	6.9E-7
G	Early/isolation failure, containment failure prior to/at vessel failure, up to 10% volatiles released	1.4E-7	2.0E-8
B	Containment bypassed, < 0.1% volatiles released	2.3E-7	2.2E-7
D	Containment bypassed, up to 10% volatiles released	0	1.2E-6
T	Containment bypassed, > 10% volatiles released	0	6.5E-7

1.4.3 CONCLUSION

The IPEEE effort has met the objectives of the NRC for Generic Letter 88-20, Supplement 4 which were essentially: 1) to develop SCE's understanding of plant specific responses to severe accidents, and 2) to implement changes where indicated.

The majority of the modeling, quantification, and prioritization of core damage and significant release sequences associated with the IPE were performed by in-house personnel, thus assuring that a detailed appreciation of severe accident behavior was developed within SCE.

The IPEEE identified several plant and procedural changes that provide substantive and cost effective risk benefit. These changes were included in all modeling and are reflected in the final results. These changes include:

- improvement in the reliability of cross-connecting Units 2 and 3 to allow a unit's emergency diesel generators to supply power to the other unit in the event the other unit has a station blackout (improved 4kV power availability)

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- strengthening of ammonia tank supports (removes ammonia spill hazard)

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- removal of floor grating surrounding auxiliary feedwater (AFW) valve actuators (allow valve movement without spatial interaction with surrounding grating)

  STATUS: Implementation by December 30, 1995.

- removal of concrete plug surrounding Unit 2 diesel generator fuel oil transfer piping (2) (improves piping's seismic capacity)

  STATUS: Implementation by December 30, 1995.

- fastening adjacent electrical cabinets/panels together (prevent interactions and relay chatter)

  STATUS: Implementation by March 31, 1996.

- stabilizing two light fixtures that interact with electrical cabinets

  STATUS: Implementation by December 30, 1995.

- modification of several station procedures to address internal fire findings. These include:

  a. an administrative change to procedure SO23-13-2 ("Shutdown from Outside the Control Room") to allow operators to utilize offsite power in the event that the reserve auxiliary transformers are not inadvertently tripped by fire-induced damage to control room panel 2/3CR-63.

     STATUS: Implementation by the end of the Cycle 9 refueling outage.

  b. an administrative change to procedure SO23-13-21 ("Fire") to allow operators to recover power to the 4 kV switchgear by disconnecting power to the diesel generator feeder breaker and reclosing the offsite power breaker on the switchgear.

     STATUS: Implementation by the end of the Cycle 9 refueling outage.

  c. an administrative change to alarm response procedure SO23-15-60.A1 (Annunciator Panel 60A, Emergency HVAC) would allow operators to use air ducting and gas driven fans to prevent room heat-up. This enhancement also reduces the risk due to seismic and internal events.

     STATUS: Implemented.

1.5 REFERENCES

1-1 "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, USNRC, November 23,1988.

1-2 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement 4, USNRC, June 28, 1991.

1-3 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, " NUREG-1407, Final Report, USNRC, June 1991.

1-4 "PRA (Probabilistic Risk Assessment) Procedures Guide," Volumes I and II, NUREG/CR-2300, USNRC, January 1983.

1-5 "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," NUREG/CR-4840, USNRC, November 1990.

1-6 "Seismic Hazard at San Onofre Nuclear Generating Station," Final Report, Risk Engineering, Inc., August 1995.

1-7 "Fire-Induced Vulnerability Evaluation (FIVE)," EPRI TR-100370, April 1992.

2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

In response to GL 88-20, Supplement 4, SCE performed a probabilistic risk assessment of SONGS 2/3 involving an integrated analysis of plant and system response to externally initiated events in order to evaluate and quantify plant core damage frequency and evaluate containment performance [2-1]. The analysis is based on the SONGS 2/3 Cycle 7 plant design with modifications described in Section 7.1.

2.2 CONFORMANCE WITH GENERIC LETTER 88-20 AND SUPPORTING MATERIAL

The IPEEE of SONGS 2/3 as summarized in this submittal conforms to the NRC guidance contained in GL 88-20, Supplement 4 and NUREG-1407 [2-2]. SCE submitted a letter dated December 23, 1991, to the NRC outlining the proposed SCE IPEEE Program Plan for SONGS 2/3.

The criteria used in selecting important severe accident sequences were in accordance GL 88-20, Supplement 4. Documentation of examination results was maintained in a traceable manner under in-house document control as required. This IPE report contains the information required by Appendix 4 of GL 88-20, Supplement 4, and follows the format described in Table C.1 of NUREG-1407.

2.3 GENERAL METHODOLOGY

The IPEEE includes three risk analyses: 1) seismic, 2) internal fires, and 3) high winds, floods and other external hazards. The seismic and internal fire analyses include a Level I (Front-End) analysis of core damage frequencies and a Level II (Back-End) analysis of phenomena affecting containment behavior and the release of radionuclides to the environment. The high winds, floods and other external hazards analysis includes a progressive screening process to confirm the plant conformity to 1975 Standard Review Plan Criteria (per NUREG-1407) [2-7].

As required in NUREG-1407, the SONGS 2/3 seismic analysis is a full scope seismic PRA. This includes a plant-specific seismic hazard characterization, plant systems and structures response analyses, extensive plant walkdowns, fragility evaluations of systems, structures and components, relay chatter evaluations, human reliability analysis, plant system and sequence analysis, and containment systems analysis. The methods used in the seismic PRA are consistent with NUREG/CR-2300, and NUREG/CR-4840 [2-3, 2-4]. A detailed seismic analysis methodology discussion is provided in Section 3.0.

In conformance with NRC GL 88-20, Supplement 4, and NUREG-1407, the internal fire analysis used a combination of the two NRC-approved approaches, FIVE and fire PRA. The EPRI FIVE [2-5] methods were used for progressive screening of most fire compartments, and more detailed COMPBRN fire modeling [2-6] and PRA methods were used for the analysis of non-screened compartments. A detailed internal fire analysis methodology discussion is provided in Section 4.0.

As described in Generic Letter 88-20 and NUREG-1407, the SONGS 2/3 IPEEE assessed the impact of high winds, floods and other hazards on the two units using a screening approach to ensure that all requirements of the 1975 Standard Review Plan were still met [2-7]. A special walkdown was performed to verify that the plant conditions with respect to external events documented in the UFSAR had not changed. A detailed discussion of the methodology is provided in Section 5.0.

2.3.1 APPLICABILITY OF RESULTS TO BOTH UNITS

For the seismic risk analysis, walkdowns were performed for each unit to assess any differences in plant seismic behavior. Walkdowns identified that the diesel generator fuel oil transfer system piping for Unit 2 was different than Unit 3 in that the Unit 2 piping was encased in concrete at the building's exterior wall preventing pipe movement during a seismic event. A seismic event may rupture the pipe and prevent long term DG operation. After the Unit 2 diesel generator fuel oil transfer piping tunnel is modified to remove the concrete (implementation discussed in Section 7.1), there will be no risk significant seismic differences between Unit 2 and 3, and therefore, this analysis is applicable to both units.

With respect to fire risk, two fire compartments in Unit 3 [3-PE-45-3A & 3-SE-(-15)-136] were different than the corresponding compartment at Unit 2. All other compartments are essentially identical between units. The FIVE analysis was performed by including the fire risk of the two unique Unit 3 compartments with the Unit 2 compartments. The resulting FIVE analysis is conservative and bounds both units.

The response of SONGS 2 and 3 to other external hazards are identical. Therefore, the examination results for seismic, fire, and other external hazards are applicable to both SONGS Units 2 and 3.

2.3.2 VULNERABILITY DEFINITION

As defined in the SONGS 2/3 IPE and used in the IPEEE, the definition of a severe accident vulnerability is as follows:

A vulnerability in a PWR is a plant feature which contributes a disproportionately large percentage to either core damage or significant release probabilities which are in turn significantly higher than those of an average PWR.

This definition is applicable for the seismic, internal fire, and other hazards analysis.

2.4 INFORMATION ASSEMBLY

The information gathered during the SONGS 2/3 IPE analysis provided the primary source of information for the SONGS 2/3 IPEEE. The procurement of additional information to support the IPEEE included review of plant-specific historical hazard data (including seismic, high winds, transportation and other external hazards), other external events PRA studies, and generic fragility information. SONGS 2/3 piping and instrumentation drawings, mechanical line isometrics, electrical elementaries, electrical one-line diagrams, and operations & maintenance procedures provided additional information. Extensive walkdowns for seismic, internal fire, and high winds, floods, and other hazards events served a large and valuable source of information.

All major work products (i.e., fire hazards and location analyses, seismic hazards characterization, seismic equipment and relay lists, relay chatter evaluations, fragility analyses, containment response analyses, etc.) were subjected to a multi-disciplinary review by consultants and SCE personnel to ensure that the plant, systems, and procedures were reflected accurately.

2.5 REFERENCES

2-1 "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f)," G.L. 88-20, Supplement 4, June 1991.

2-2 "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, June 1991.

2-3 "PRA Procedures Guide", NUREG/CR-2300, January 1983.

2-4 "Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150," NUREG/CR-4840, November 1990.

2-5 "Fire-Induced Vulnerability Evaluation (FIVE)", EPRI TR-100370, April 1992.

2-6 "COMPBRN IIIE: An Interactive Computer Code for Fire Risk Analysis," EPRI Report NP-7282, May 1991.

2-7 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants -- LWR Edition," NUREG-75/087, December 1975.

3. SEISMIC ANALYSIS

This section provides a description of the methodology used to perform the seismic analysis for SONGS 2/3, and a synopsis of the significant results for each portion of the analysis. The seismic analysis fulfills the objectives of the IPEEE, and provides a systematic examination to identify any plant-specific vulnerabilities to severe accidents initiated by seismic events. The organization of this section is:

Appendices contain the detailed hazard information, seismic equipment list, and dominant cut sets. The following table provides a cross-reference between the NUREG-1407 Standard Table of Contents and this submittal:

NUREG-1407		SONGS 2/3 Submittal
3.0	Methodology Selection	3.0
3.1	Seismic PRA	3.1
3.1.1	Hazard Analysis	3.1
3.1.2	Review of Plant Information and Walkdown	3.2 (Plant Information)
		3.3 (Walkdowns)
3.1.3	Analysis of Plant System and Structure Response	3.4
3.1.4	Evaluation of Component Fragilities and Failure Modes	3.5
3.1.5	Analysis of Plant Systems, Sequences	3.6
3.1.6	Analysis of Containment Performance	3.7
3.2	USI A-45, GI-131, and Other Seismic Safety Issues	3.8

3.0 METHODOLOGY SELECTION

In conformance with NRC Generic Letter 88-20, Supplement 4, and NUREG-1407, the seismic analysis used the NRC-approved seismic PRA approach. The overall process is depicted in Figure 3.0-1, and the major steps are briefly described below.

Hazard Analysis

A plant-specific seismic hazard analysis was performed specifically for this seismic PRA, utilizing state-of-the-art techniques and data. SCE initially presented these techniques along with the methods to develop seismic component and structural fragilities to the NRC staff on January 27, 1994. The preliminary results of the seismic hazard analysis were later presented at a follow-up meeting with the NRC on November 27, 1994.

The result of the analysis is a description of the annual frequency of exceedance of various ground motion levels (accelerations) at SONGS and the associated uncertainty. The study considered multiple interpretations about the causes and physical characteristics of potentially active faults and area sources in order to characterize seismic hazard uncertainty. Similarly, uncertainties in the ground motion attenuation equations were propagated through the hazard analysis. The result was a suite of hazard curves (at 6 frequencies), and their associated weights, which represent the seismic hazard at the site and the associated uncertainty. These hazard curves were then combined to determine the mean hazard curve, which was used for the baseline analysis, as permitted by NUREG-1407. Extensive sensitivity studies were performed for seismicity parameters such as slip rates, depth, and maximum magnitude, and for attenuation equations. The hazard study was performed by three teams, with an expert review panel providing feedback during the entire project. Section 3.1 provides more detail on the methodology, input data, and results.

Plant Information and Selection of Systems and Equipment

A comprehensive approach was used to identify systems and equipment that can provide safe shutdown of the reactor, and maintain a safe, stable state after a beyond design basis earthquake. A seismic equipment list (SEL) was developed which includes the plant systems and components providing safety functions to prevent core damage, as well as the structures, equipment, and actuation components necessary for the functions of containment integrity, containment pressure suppression, containment heat removal, containment radioactivity removal, and containment isolation.

A plant-specific procedure was followed which used the internal events IPE as the initial basis for the identification of the appropriate safety functions and systems, and the required equipment. However, several additional steps were used to identify equipment which was not in the IPE, but which would be important during and after an earthquake. For example, some components such as heat exchangers and filters which maintain piping system boundary integrity and prevent flow diversion were added to the SEL. Other components not explicitly in the IPE but added to the SEL are items such as electrical panels and cabinets which house SEL items. The relevant emergency operating procedures were reviewed and discussed with the training staff to verify that equipment and instrumentation used in the procedures, and considered critical to safe shutdown, were included in the SEL. Particular attention was placed on equipment important to containment performance, including the potential for interfacing systems LOCA, containment bypass, and containment isolation and actuation. A special effort was made to include equipment which could cause seismic-induced fires or floods, or releases of toxic or flammable gases.

A comprehensive list of relays associated with the control, actuation, and instrumentation of the above equipment was generated separately, and used for the relay chatter and seismic capacity evaluation task. A separate procedure was used to guide this relay list development, with a series of checks on the process to ensure completeness. The relay list includes process switches such as temperature and pressure switches.

The overall results of this task were the seismic equipment list (SEL), which was used to guide the seismic capacity walkdowns, and the seismic relay list (SRL), which was used to guide the relay chatter evaluation as well as identify cabinets for the seismic capacity and relay walkdowns.

Walkdowns

One of the most important tasks in the seismic PRA was the systematic walkdown of components on the SEL. The purpose was to identify equipment vulnerabilities in either the component load path or anchorage, potential seismic failure/falling and proximity interactions, and potential flooding or fluid spray interactions, including multiple concurrent flooding sources when credible. The walkdowns were performed by teams of experienced seismic capability engineers, using the EPRI NP-6041 procedures and worksheets. Extensive documentation was taken and incorporated into a seismic walkdown database. Based on these walkdowns, associated seismic qualification and anchorage calculations, many of the SEL items were screened at this stage as having high seismic capacity. Items which could not be screened required seismic fragility calculations.

Analysis of Plant System and Structural Response

In order to calculate the seismic demand which could be placed on structures and components from a beyond-design-basis earthquake, realistic estimates of structural response to seismic events were developed. In general, floor response spectra and structure member forces developed for the plant design basis are considered to be conservatively biased. Therefore, a state-of-the-art median-centered seismic response analysis was performed, with particular consideration of soil-structure interactions (SSI) since SONGS buildings are located on a deep soil site. The overall methodology followed these steps:

1. Specification of the free-field ground motion (from the seismic hazard analysis)
2. Development of the soil models
3. Calculation of the foundation impedance functions and wave scattering effects
4. Determination of the fixed-base dynamic characteristics of the structures
5. Performance of the SSI analysis to calculate the response of the coupled soil-structure system

The two main results were the estimated median structure forces and the variability about the median for all structures of interest, and the probabilistic floor response spectra in these structures. These were then used for the structure fragility analysis and the equipment fragility analysis.

Evaluation of Component Fragilities and Failure Modes

For those structures and components that were not screened out based on the seismic capacity walkdowns, progressively more detailed calculations were performed to estimate the seismic capacity of each component. In essence, the factors of safety, conservatism, and over-design that are common in the seismic design, analysis, construction, and installation of structures and components are estimated, and a realistic estimate of the ability of a structure or component to withstand an earthquake is calculated. This capacity is expressed in the form of a family of fragility curves, with parameters for the median capacity, and the random and modeling uncertainties (β_R, β_U). This provides a realistic estimate of the probability of failure of the component (or structure) at each level of ground acceleration. Relays and switches were included in these fragility calculations.

Analysis of Plant Systems and Sequences

The analysis of plant systems and potential accident sequences was similar to the internal events IPE, and used many of the same models and data. The primary model difference is that a seismic event tree was developed to delineate the potential combinations of seismic-induced failures, and resulting seismic scenarios, which were termed "seismic damage states." Traditional event tree techniques were used to identify each of the top seismic-induced events, and to formulate the nodal branching logic. The frequencies of these seismic damage states were quantified by convolving the SONGS site-specific mean earthquake hazard curve with the structure and equipment seismic fragility curves. This quantification included dependent and correlated failures, and appropriate success states. Some seismic-specific operator actions were included in the analysis. For those scenarios that required additional non-seismic failures to occur to result in core damage, the IPE internal events model (event trees and fault trees) was used to develop conditional core damage probabilities, with appropriate changes given the seismic damage state. These calculations incorporate random failures of equipment and operator actions. The overall frequency of seismic-induced core damage is then quantified from these intermediate results by adding up the individual scenarios.

The results are expressed in terms of dominant seismic sequences, dominant contributors, and contribution of the various acceleration ranges to core damage frequency. Sensitivity studies were performed for certain key issues, and uncertainty analysis was performed for seismic hazard and fragility curve uncertainties.

Containment Performance

A number of containment performance related structures, systems, and components were evaluated to determine any unique containment performance issues, particularly with respect to the potential for containment bypass or early, large releases to the environment. The methodology used extended plant damage state event trees and fault trees, containment event trees, and source term grouping logic trees to fully quantify the dominant seismic scenarios, and determine the impacts on containment performance and source terms.

USI A-45, GI-131, and Other Seismic Safety Issues

In accordance with the IPEEE request, USI A-45 Decay Heat Removal, GI-131 In-core Flux Mapping Seismic Interaction, seismic-induced fire and flood interactions, and other seismic safety issues were specifically identified and discussed.

A summary section is provided to list any potential plant-specific vulnerabilities, and document the status of planned plant modifications.

The following subsections provide more detail on the methods used, and the results and insights.

3.1 SEISMIC HAZARD ANALYSIS

The probabilistic hazard of earthquake-induced ground shaking at the San Onofre Nuclear Generating Station (SONGS) was investigated for the IPEEE. The results were used to guide decisions regarding seismic safety and levels of seismic evaluation at the facility. The express purpose of the study was to follow the methodology developed by several recent studies of seismic hazard at nuclear facilities in the U.S., so that the same insights gained from other studies regarding seismic hazards and risk mitigation can be used at SONGS. These other studies make explicit representation of the uncertainty in seismic hazard caused by multiple, alternative hypotheses of the causes and characteristics of earthquakes.

The design of the SONGS study has been to use one team (Geomatrix Consultants, Inc.) to derive seismic sources, a separate team (Woodward-Clyde Consultants, Inc.) to derive ground motion attenuation equations, and a third team (Risk Engineering, Inc.) to perform historical seismicity analysis, integrate the study, and compute the seismic hazard. In addition, a peer review team consisting of:

Dr. Norman Abrahamson -- Consultant
Professor Keiiti Aki -- University of Southern California
Professor Clarence Allen -- California Institute of Technology

has been involved from the beginning of the project to review ongoing studies and comment on preliminary results and reports. This arrangement has provided a strong scientific and engineering basis for the interpretations utilized for SONGS.

Recent studies of seismic hazard in the central and eastern United States (CEUS) have been completed by the Electric Power Research Institute, funded by the Seismicity Owners Group (EPRI/SOG) [3-2], and by the Lawrence Livermore National Laboratory (LLNL), funded by the U.S. Nuclear Regulatory Commission [3-3]. These studies represent major efforts to characterize the seismic hazard for nuclear power plants in the CEUS and use the most recent, up-to-date understandings of seismicity and ground motion relations to the region. Also, Pacific Gas and Electric Co. [3-4] performed a major study of seismic hazard at the Diablo Canyon Power Plant in California specifically to incorporate alternative hypotheses on tectonics, seismicity, and ground motion values into the decision process regarding seismic hazards.

A general description of faults, area sources, and parameters that expresses quantitative uncertainties in interpretations (although such a description is being developed by the Southern California Earthquake Center) is not available for southern California. Therefore, a description of this type was developed by Geomatrix Consultants, Inc., treating earthquake occurrences both on faults and in area sources (where specific faults have not been identified). Following the methodology of the other studies indicated above, multiple interpretations are considered for faults and area sources in order to characterize uncertainty in the seismic hazard that results from uncertainty in earthquake characteristics.

SONGS is located at latitude 33.369° north and longitude 117.554° west. Structures at the site are founded on stiff soils overlying bedrock. Mathematical functions describing earthquake ground motion dependence on magnitude and distance in southern California were derived by Woodward-Clyde Consultants, Inc. These functions were used to describe ground motion, its randomness, and its uncertainty. Consistent with other state-of-the-art seismic hazard studies, these functions are used to derive the distribution of seismic hazard for spectral accelerations (S_a) at frequencies from 0.5 Hz and above. Hazard spectra are shown to be constant to demonstrate typical spectral amplitudes and shapes that are appropriate for earthquake ground motions of interest.

3.1.1 METHODOLOGY

This section describes the methodology used to calculate seismic hazard in a general way. Specific inputs to the methodology are described in subsequent sections.

State-of-the-art seismic hazard studies calculate ground-motion exceedence probabilities using earth-science hypotheses about the causes and characteristics of earthquakes in the region being studied. Scientific uncertainty about the causes of earthquakes and about the physical characteristics of potentially active tectonic features lead to uncertainties in the inputs to the seismic hazard calculations. These uncertainties are quantified using the tectonic interpretations developed by earth scientists familiar with the region. These experts evaluate the likelihood of the seismic hazard associated with alternative tectonic features and with alternative characteristics of these potential sources.

These and other uncertainties (e.g., ground motion attenuation equations) are carried through the entire analysis. The result of the analysis is a suite of hazard curves and their associated weights. These curves quantify the seismic hazard at the site and its uncertainty.

3.1.1.1 Basic Seismic Hazard Model

The methodology to calculate seismic hazard at a site is well established in the literature [3-5,3-6,3-7,3-8,3-9]. Calculation of the hazard requires specification of three inputs:

1. Source geometry: the geographic description of the seismic source. A seismic source is a portion of the earth's crust, associated with a fault, with a concentration of historic seismicity, or with a specific tectonic feature (other than a fault), that may be capable of producing earthquakes. Source geometry determines the probability distribution of distance from the earthquake to the site: $f_R(r)$.

2. Seismicity: the rate of occurrence v_i and magnitude distribution $f_{M(i)}(m)$ of earthquakes occurring in each source i. Magnitude is usually characterized by the moment magnitude scale **M** in California and the Rocky Mountain region, and by the body-wave magnitude m_b in the central and eastern U.S. (CEUS).

3. Attenuation functions: a relationship that allows the estimation of ground motion at the site as a function of earthquake magnitude and distance, incorporating known effects of surficial soils on seismic motions.

These inputs are illustrated in Figure 3.1-1, parts a through c. Figure 3.1-1a shows the geometry of a seismic source. From the source's geometry, $f_{R(i)}(r)$ can be derived. The density function on magnitude $f_{M(i)}(m)$ is often specified as the doubly truncated exponential distribution for area sources and the characteristic magnitude distribution for faults, as illustrated in Figure 3.1-1b [3-10]. The characteristic magnitude distribution has a "hump" for the seismicity in the higher magnitude range. Seismicity for a source with the exponential magnitude distribution is completely specified by the minimum magnitude m_o and parameters a and b. Parameter a is a measure of seismic activity, b is a measure of relative frequency of large versus small events, and log[$v_i f_{M(i)}(m)$] is proportional to $a + b\,m$ for $m_o < m \leq m_{max}$. For the characteristic magnitude distribution, it is necessary to additionally specify the "characteristic" part of the distribution, i.e., the magnitude range of earthquakes that act in a characteristic way and the annual rate of occurrence of magnitudes in that range.

The ground motion is modeled by an attenuation function, as illustrated in Figure 3.1-1c. Attenuation functions are usually of the form $\ln[A] = f(M,R) + \epsilon$, where A is ground-motion amplitude, M is magnitude, R is distance, and ϵ is a random variable that represents scatter. The attenuation function is used to calculate

$$G_{A/m,r}(a^*) = P[A > a^*/m,r]$$

which is the probability that the ground-motion amplitude is larger than a^*, for a given M and R. The seismic hazard over all sources is calculated as a summation:

$$\frac{P[A > a^* \text{ in time } t]}{t} = \sum v_i \iint P[A > a^*|m,r]\, f_{M(I)}(m)\, f_{R(I)(r|m)}\, dm\, dr \qquad (3\text{-}1)$$

in which the summation is performed over all seismic sources I and in which the probability is calculated per unit time.

3.1.1.2 Tectonic and Seismicity Interpretations

The specification of potential sources of future earthquakes is the first step in the evaluation of earthquake hazards. Seismic sources indicate where earthquakes may occur. Analysis of historical seismicity within those defined sources indicates the probabilities of occurrence and characteristics of future earthquakes (i.e., a magnitude distribution is derived from historical data within the source once the source is defined).

A seismic source is by definition a fault or area with a single probability of being active, a single magnitude distribution, and a single distribution for maximum magnitude. Within a seismic source the seismicity is usually taken to be spatially homogenous (i.e., earthquakes are assumed to be equally likely to occur at any location within the source). Some studies (e.g., the EPRI/SOG study) use spatially-varying seismicity, but this generalization was not adopted.

In general, seismic sources are defined based on faults, tectonic features, or other evidence (including, in some cases, merely a spatial cluster of historical seismicity). Because of this derivation there is, conceptually, some causal association of earthquakes within a source: they are releasing crustal stresses of the same orientation and amplitude, and/or they are caused by slip on faults with the same general depth, orientation, and sense of slip. Because of these similarities the delineation is consistent with the seismic source definition with regard to maximum magnitude and probability of activity.

3.1.1.3 Seismicity Parameters

Seismicity parameters for earthquake sources are estimated using the rate of tectonic slip for active faults, and using historical seismicity for area sources. The rate of slip on faults is important because multiple methods of estimation can be applied, including measured offsets of datable horizons, crustal strain measurements or inferences, mechanistic tectonic block models of crustal plates, and paleoseismicity studies. For area sources, earthquake catalogs are analyzed to collect all seismic events that have occurred within each source. For each magnitude level, periods of completeness are picked and the rate of occurrence for that magnitude level is calculated as the number of events divided by the time of complete observation. These data are then fit using the maximum-likelihood procedure to obtain estimates of a and b [3-11].

When the characteristic magnitude distribution is used, the rate of occurrence of events with the characteristic size must generally be estimated using data other than historical seismicity. This is the case because there are few places in the U.S. where a sufficient number of cycles of seismicity have been observed historically to calculate a rate of characteristic events from observations. For some faults (e.g., the San Andreas), paleoseismic evidence gives some indication of the rate of occurrence of the characteristic earthquakes.

Maximum magnitude distributions are estimated using a combination of techniques [3-12, 3-13). Among these are fault length-magnitude relations, comparison with other regions of similar characteristics, consideration of geophysical characteristics that relate to m_{max}, and consideration of the amount of information known about the region under consideration. Ultimately the choice of m_{max} distribution should be made by analysts familiar with the region.

The choice of minimum magnitude m_o is based on the characteristics of small earthquakes (i.e., on how damaging are the ground motions associated with these earthquakes), analysis of structural response for the facilities being studied, and field observations of structural performance during low-intensity ground motions. Convention in current studies is to use a moment magnitude of 5.0 for m_o, which is supported by studies of the damage ability of ground motions from small magnitude earthquakes References [3-14, 3-15]. These studies were made for generic nuclear power plant structures and equipment, and there is no reason to believe that they would not be applicable to SONGS.

3.1.1.4 Ground Motion Attenuation Equations

Equations estimating seismic ground motion are required for the seismic hazard calculations. These are selected using ground motion studies conducted in the region, available strong motion and seismological data, and inferences from characteristics of earthquakes. Equations are selected for all measures of interest for the study, which are spectral accelerations (S_a) corresponding to 5% damping for frequencies of 0.5 Hz and above. Ground motion estimates exhibit randomness and the standard assumption in seismic hazard analyses is to characterize the randomness using a log normal distribution with a specified standard deviation of ln[ground motion]. Typically, the value of $\sigma_{\ln[\text{ground motion}]}$ varies as a function of structural frequency and it may also vary with magnitude of the earthquake.

3.1.1.5 Calculations

Equation 3-1 is formulated using the assumption that earthquakes (most particularly, successive earthquakes) are independent in size and location. In all seismic hazard applications, primary interest is focused on computing probabilities for high (rare) ground motions. As a result, the probability of two exceedences in time t is negligible. The same argument holds when considering hazard at a site from multiple sources. Thus, the summation on the right side of Equation 3-1 -- which is the rate of earthquakes with $A > a^*$ -- is a good approximation to the probability of exceeding amplitude a^* in time t. This is why Equation 3-1 is an approximation (but an accurate one), not a strict equality.

The calculation of hazard from all sources is performed for multiple values of a^* in order to generate the hazard curve, which gives the annual probability of exceedence as a function of a^*. This calculation is performed in the current study for six different measures of ground motion: S_a at the frequencies of 25, 10, 5, 2.5, 1, and 0.5 Hz, all at 5% damping.

3.1.1.6 Treatment of Uncertainty

State-of-the-art seismic hazard studies distinguish between two types of variability: randomness and uncertainty. "Randomness" is the probabilistic variability that results from natural physical processes. The size, location and time of the next earthquake on a fault and the details of the ground motion are examples of random events. In concept, these elements cannot be predicted even with collection of additional data, so the randomness component of variability is irreducible. The second category of variability is "uncertainty" which is the statistical or modeling variability that result from lack of knowledge about the true state of nature. In principle, this variability can be reduced with the collection of additional data.

These two types of variability are treated differently in advanced seismic hazard studies, as follows. Integration is carried out over probabilistic variabilities to get a single hazard curve (as indicated by Equation 3-1). Modeling uncertainties are expressed by multiple assumptions, hypotheses, or parameter values.

There are uncertainties associated with each of the three inputs to the seismic-hazard evaluation, as follows:

- Uncertainty about seismic sources and faults (i.e., what tectonic features in a region are actually earthquake sources) arises because there are multiple hypotheses about the causes of earthquakes and because there is incomplete knowledge about the physical characteristics of tectonic features. Uncertainty may also arise about the geometry of a seismic source.

- Uncertainty in seismicity is generally divided into uncertainty in maximum magnitude and uncertainty in seismicity parameters v and b. Uncertainty about m_{max}, the maximum magnitude that a given source can generate, arises for the same reasons described above. Estimates of m_{max} are obtained from physical characteristics of the source and from historical seismicity. Uncertainty in seismicity parameters v and b arises from statistical uncertainty and from uncertainty about the accuracy of various catalogs of historical seismicity available with which to estimate parameters. For the characteristic magnitude distribution, additional uncertainties are the magnitude range of the characteristic event and its annual rate of occurrence.

- Uncertainty in the attenuation functions arises from alternative hypotheses about the ground motion characteristics associated with earthquakes. This uncertainty often is large, particularly in areas where few direct recordings of strong motion are available.

These multiple interpretations are used to calculate alternative seismic hazard values according to Equation 3-1, resulting in a suite of hazard curves. The weight assigned to each seismic hazard curve is calculated from the probabilities given to each of the uncertain inputs used to calculate it; the final weight is calculated as the product of the probabilities of the input variables. From the suite of hazard curves, each with an associated weight, fractile curves or a mean seismic hazard curve are derived.

In order to organize and display the multiple hypotheses, assumptions, parameter values and their possible combinations, a logic tree approach is used in this study. Logic trees are a convenient means to express alternative interpretations and their probabilities. Each node of the logic tree represents one source of uncertainty. The branches emanating from one node represent possible alternative values of a parameter. The probability assigned to a branch represents the likelihood of the parameter value associated with that branch, and these parameter values (and probabilities) may depend on values of the preceding parameters.

The logic tree in Figure 3.1-2 illustrates the treatment of parameter uncertainty. There is one hazard curve associated with each terminal node; this hazard curve corresponds to certain sources being active, each active source having a certain m_{max} and certain seismicity parameters, and a certain attenuation function being the "correct" attenuation model. The probability associated with that end branch is the product of the probabilities of all branches traversed to reach that end branch.

Logic trees are a convenient way of organizing the uncertainties incorporated into a seismic hazard analysis and of documenting them as well.

3.1.1.7 Summary of Methodology

Probabilistic seismic hazard analysis requires as input a delineation of seismic sources, a specification of seismicity characteristics for those sources consisting of magnitude distributions and associated parameters, and a selection of ground motion attenuation equations for the region of interest. In concept all possible earthquakes in the region are modeled, as are the associated ground motions. Uncertainties in active faulting, areal sources, characteristics of seismicity, and ground motion are incorporated explicitly as multiple alternative hypotheses. The effects of these uncertainties are represented as uncertainty in the hazard curves, and sensitivity studies show the influence of each input uncertainty on the resulting calculated hazards. Thus the hazard analysis is an overall methodology that can represent both randomness and uncertainty in earthquake occurrences, characteristics, and ground motions, for the purpose of decision-making regarding seismic risk mitigation.

3.1.2 SUMMARY OF INPUT DATA

3.1.2.1 Seismic Sources

This section summarizes the seismic sources used for calculation of the seismic hazard at SONGS. The seismic sources were delineated by Geomatrix Consultants, Inc., and are documented in detail in Reference [3-1].

Figure 3.1-3 shows the active faults identified as possible sources of earthquakes in southern California, and Figure 3.1-4 indicates the area sources in the vicinity of SONGS that were used to represent seismicity that occurs away from known faults. All of these sources were investigated to determine their possible contribution to the seismic hazard at SONGS.

To analyze historical seismicity, Geomatrix defined corridors around each fault, as shown in Figure 3.1-5. This allowed historical seismicity to be collected around each fault, for comparison of observed rates of activity to rates predicted from fault slip rate and the characteristic earthquake model. These corridors also allowed seismicity to be assigned to the identified faults, so that the remaining historical seismicity could be modeled using the area sources (Figure 3.1-4).

3.1.2.2 Seismicity Parameters

To derive seismicity parameters for the faults and area sources shown in Figures 3.1-3 and 3.1-4, three earthquake catalogs were used. The first is the catalog of Ellsworth [United States Geological Survey (USGS)], who studied all events with **M** > 6 and determined epicentral locations and magnitude estimates. The second catalog was obtained from the National Oceanic and Atmospheric Administration, and includes earthquakes in southern California prior to 1932. The third catalog was obtained from the California Institute of Technology; it includes events in southern California in 1932 and later years. These three catalogs were combined, with the Ellsworth locations and magnitudes given preference over those from the other two catalogs because he has studied earthquakes with **M**>6 in California and synthesized location and magnitude estimates from other sources. A plot of the epicenters in the catalog is shown in Figure 3.1-6. Within each catalog, instrumental magnitudes and intensities were used to characterize each event. Each of these were converted to a consistent magnitude measurement **M**, or moment magnitude. The Gutenberg and Richter method [3-19] was used to convert MMI to **M**: $\mathbf{M} = 1 + 2/3$ MMI. The instrumental magnitudes were converted to **M** using equations derived from a graph by Boore and Joyner [3-18]. A total of 13,844 events with $\mathbf{M} \geq 3$ are present in the catalog.

Seismic hazard is calculated for mainshocks only (this is a common element of the EPRI/SOG, LLNL, and PG&E studies, for example). The use of mainshocks only is standard in seismic hazard analyses and allows these results to be compared on a consistent basis to seismic hazard results produced for other nuclear plant sites. To identify aftershocks and other dependent events, the algorithm of Reasenberg (1985) was adopted and applied to the southern California catalog. This resulted in 7,051 events being identified as aftershocks or other dependent events; these are plotted in Figure 3.1-7. The remaining 6,793 mainshocks are shown in Figure 3.1-8.

For both the fault corridors and the area sources described in the Section 3.1.2.1, an analysis was conducted to determine rates of activity and b-values for each seismogenic zone. This analysis proceeded with the following steps:

1. For each seismogenic zone, determine earthquakes that fall within the boundaries of that zone.

2. For specific magnitude ranges, adopt the times of complete reporting described by Engdahl and Rinehart [3-17], and determine the number of earthquakes observed in that magnitude range over the time of complete reporting.

3. Use the maximum-likelihood procedure of Weichert [3-11] to calculate an activity rate and *b* value for seismicity in the zone.

For these calculations, preliminary estimates of the upper-bound magnitude were used; this is sufficient because the calculated activity rates and *b*-values are insensitive to the choice of $\mathbf{M}_{max}$ value.

The calculated historical rates of activity were used in two ways. For the faults, seismicity within the fault corridors was compared to the rate of activity predicted using fault slip rate, as determined by Geomatrix [3-1]. Figure 3.1-9 indicates the historical seismicity and maximum-likelihood fit for Newport-Inglewood-Southern California Offshore Zone of Deformation (SCOZD) fault zone. Reference 3-1 contains the figures for the other faults that were considered.

It is important to note the methodology used to obtain the seismicity in the blind thrust regions. Los Angeles Basin Sources A and B seismicity counts were obtained after the extraction of events by other regions. No detailed study was done to correlate earthquakes to fault or blind thrust.

Also shown on the figure is the predicted seismicity rates using the fault slip rates, as documented in Reference [3-1] by Geomatrix. This predicted seismicity assumes a characteristic magnitude model. The comparison indicates that historical seismicity is generally within the range of the predicted rates of activity derived from the slip rate model; some faults indicate higher rates of activity historically, and some lower.

For area sources, the historical seismicity was used to estimate rates of activity and *b*-values for seismic hazard calculations. For these area sources the seismicity associated with the fault corridors was first removed, so that it would not be double counted in estimating the rates of activity.

Table 3.1-1 summarizes the mean seismicity parameters for faults and area sources, as derived from slip rate for the faults and from historical seismicity for the area sources. Uncertainty in rates, *b*-values, and $\mathbf{M}_{max}$ were incorporated into the hazard analysis; details of the uncertainty distributions for faults are documented in the Reference [3-1] and the area sources are shown in Table 3.1-2.

3.1.2.3 <u>Ground Motion Attenuation Functions</u>

This section summarizes the ground motion attenuation equations used to estimate ground shaking at SONGS as a function of earthquake magnitude and distance. Details of the functions by Woodward-Clyde Consultants are given in Reference [3-1].

Five horizontal attenuation equations were evaluated based on comparisons of the five equations with strong motion data. All five equations estimate ground motion at the surface of a stiff soil column, and therefore are appropriate to use directly to estimate seismic hazard for SONGS. The equations are identified in Table 3.1-3.

Weights were assigned to the equations based on how well they fit a strong motion data set representative of southern California. The assigned weights are indicated in Table 3.1-3, and details of the comparisons are given in Reference [3-1]. Table 3.1-3 also indicates how the standard error varies (with frequency *T* and, in some cases, magnitude **M**). Specific values of the standard error are described in Reference [3-1].

These equations were used in the seismic hazard analysis as mutually-exclusive alternatives. That is, if one equation applied to a particular fault or area source, it also applied to all others. Weights used in the hazard analysis are shown in Table 3.1-3.

Reference [3-1] shows residuals (observed minus predicted values) for the five attenuation equations and the data set used for comparison. There is some indication that the median residual tends to be negative (i.e., indicates over prediction) for spectral response. To the extent that this is the case, the resulting seismic hazard curves will be conservative.

In addition to the horizontal equations, one vertical attenuation equation (Campbell, 1990) was used to obtain a spectral shape for vertical ground motions. Details of this function are given in Reference [3-1] and the calculated spectral shapes are presented in Section 3.1.3.

TABLE 3.1-1
MEAN VALUES OF SEISMICITY PARAMETERS
FOR FAULTS AND AREA SOURCES

Source	$\nu_{5.0}$	*b*-value	M_{max}
Coronado Fault	0.0439	0.80	7.01
Elsinore Fault	0.0854	0.80	6.95
Newport-Inglewood Fault	0.0065	0.80	6.74
Newport-Inglewood-SCOZD Fault Zone	0.0236	0.80	6.86
Palos Verdes Fault	0.0385	0.80	6.81
Rose Canyon Fault	0.0132	0.80	6.66
Rose Canyon-SCOZD Fault Zone	0.0207	0.80	6.79
San Andreas Fault	0.2463	0.80	7.63
San Diego Fault	0.0103	0.80	7.14
San Jacinto Fault	0.1615	0.80	7.06
Santa Catalina Fault	0.0103	0.80	6.84
LA Basin Source A	0.0090	0.80	6.16
LA Basin Source B	0.0065	0.80	6.60
Central Los Angeles Basin source	0.0053	1.02	6.1
Central Los Angeles Basin and Peninsular Range source	0.0117	1.03	6.0
Peninsular Ranges source	0.0064	1.03	6.0
Offshore Basin source	0.0038	0.83	6.0

Note: $\nu_{5.0}$ is the annual rate of earthquakes with $M \geq 5.0$.

TABLE 3.1-2
DISTRIBUTION OF VALUES OF SEISMICITY PARAMETERS FOR AREA SOURCES

Source	$V_{5.0}$	*b-value*	$V_{5.0}$, *b*-value weight	M_{max}, weight	
Central Los Angeles Basin source	0.0221	.6935	.0278	5.5	.2
	0.0171	.6935	.1111	6.0	.4
	0.0121	.6935	.0278	6.5	.4
	0.0069	1.0229	.1111		
	0.0053	1.0229	.4444		
	0.0037	1.0229	.1111		
	0.0021	1.3523	.0278		
	0.0016	1.3523	.1111		
	0.0012	1.3523	.0278		
Central Los Angeles Basin and Peninsular Range source	0.0307	.8054	.0278	5.5	.2
	0.0257	.8054	.1111	6.0	.6
	0.0208	.8054	.0278	6.5	.2
	0.0139	1.0267	.1111		
	0.0117	1.0267	.4444		
	0.0095	1.0267	.1111		
	0.0063	1.2480	.0278		
	0.0053	1.2480	.1111		
	0.0043	1.2480	.0278		
Peninsular Ranges source	0.0236	.7309	.0278	5.5	.2
	0.0186	.7309	.1111	6.0	.6
	0.0135	.7309	.0278	6.5	.2
	0.0081	1.0297	.1111		
	0.0064	1.0297	.4444		
	0.0047	1.0297	.1111		
	0.0028	1.3285	.0278		
	0.0022	1.3285	.1111		
	0.0016	1.3285	.0278		
Offshore Basin source	0.0282	.3553	.0278	5.5	.2
	0.0182	.3553	.1111	6.0	.6
	0.0083	.3553	.0278	6.5	.2
	0.0059	.8266	.1111		
	0.0038	.8266	.4444		
	0.0017	.8266	.1111		
	0.0012	1.2979	.0278		
	0.0008	1.2979	.1111		
	0.0004	1.2979	.0278		

TABLE 3.1-3
WEIGHTS OF ATTENUATION EQUATIONS

Attenuation Relationship	Weight	Random Error
Idriss Stiff Soil Site	0.15	f(M,T)
Abrahamson Soil Site	0.20	f(M,T)
Sadigh Soil Site	0.20	f(M,T)
Boore, Joyner and Fumal Average Site Class B & C	0.25	f(T)
Campbell Soil Site	0.20	f(T) for SA f(M,T) for PGA

3.1.3 SEISMIC HAZARD RESULTS AND SENSITIVITIES

The seismic hazard results are presented in this section. These results were obtained with the computer program FRISK88M, which incorporates uncertainties in inputs to seismic hazard analyses and produces explicit hazard curves for each combination of uncertain parameters. The calculations are equivalent to the calculations performed under other modern seismic hazard studies (e.g., EPRI/SOG, LLNL, and PG&E), including the effect of fault rupture length and three dimensional geometry.

Figures 3.1-10 through 3.1-13 illustrate, by seismic fault, the annual probability of exceeding a given spectral acceleration (S_a) at a frequency of 10 Hz. As would be expected, the hypothesis of a nearby active fault (either connected to the Newport-Inglewood-SCOZD or the Rose Canyon-SCOZD faults) dominates the hazard for the larger ground motions (S_a>0.15g). At lower ground motions the San Andreas, Elsinore, and San Jacinto faults contribute most to the hazard (Figure 3.1-10). The hazards from other faults and area sources are plotted on Figures 3.1-11 through 3.1-13 to improve the readability of the plots. The area sources (Figure 3.1-14) do not contribute to much of the hazard compared to the faults.

The sensitivity to attenuation equation is shown in Figure 3.1-15 for S_a (10 Hz) for the Rose Canyon fault. This is a small contributor to the total uncertainty at low ground motions, but is a moderate contributor at the higher accelerations.

Sensitivities to seismicity parameters are illustrated as follows:

Sensitivity to slip rates:	Figures 3.1-16
Sensitivity to *b*-values:	Figures 3.1-17
Sensitivity to depth:	Figures 3.1-18
Sensitivity to $\mathbf{M}_{max}$:	Figures 3.1-19

All of these sensitivities are shown for the Rose Canyon-SCOZD fault, for S_a (10 Hz). The sensitivity to slip rate (Figure 3.1-16) indicates an important contribution of slip rate uncertainty to the hazard uncertainty. As expected, a change of a factor of 3 in slip rate (total range) results in a factor of 3 change in seismic hazard. Changes in *b*-values do not result in much change in hazard, and this is illustrated in Figure 3.1-17, where all three parametric estimates resulted in essentially the same hazard curve. The total depth of the seismogenic zone has a moderate influence on seismic hazard (Figure 3.1-18), with a 15 km depth resulting in 50% more earthquakes (and 50% more hazard) than a 10 km depth. Finally, sensitivity to $\mathbf{M}_{max}$ indicates a strong importance (Figure 3.1-19). A common seismic hazard result is seen in these plots, which is that a higher value of $\mathbf{M}_{max}$ results in lower hazard. The reason is that fault activity is characterized by slip rate. For a fixed value of slip rate, a lower value of $\mathbf{M}_{max}$ means that more earthquakes must occur (ν must be higher) to cause that slip rate, and this higher value of ν causes higher seismic hazard.

Figure 3.1-20 shows the total hazard for all faults and area sources, with uncertainties caused by uncertainties in attenuation equations and seismicity parameters. The plot is for an average S_a between 1 to 10 Hz. The uncertainty in annual probability of exceedance is lower than typical uncertainties in the central and eastern U.S. This reflects the greater knowledge about faults and activity in southern California.

The hazard results are presented in a different format in Figures 3.1-21 and 3.1-22. These are fractiles of spectra for frequencies of 25 to 0.5 Hz for annual probabilities of 1.4×10^{-4} and 1.7×10^{-6}. These probabilities were chosen because they approximately correspond to the annual probabilities of exceedence for the SSE spectrum (anchored to PGA = 0.67g) and the 2xSSE spectrum (anchored to 1.34g). Figure 3.1-23 shows mean spectra for annual probabilities of exceedence of 1.4×10^{-4} and 1.7×10^{-6}. The numerical values for these spectra are shown in Table 3.1-4.

TABLE 3.1-4
HORIZONTAL GROUND MOTIONS AT VARIOUS PROBABILITIES OF EXCEEDENCE
(Accelerations in g)

Probability		S_a (25 Hz)	S_a (10 Hz)	S_a (5 Hz)	S_a (2.5 Hz)	S_a (1 Hz)	S_a (0.5 Hz)
1E-08	mean median	2.712 2.002	5.315 3.050	6.962 4.132	6.816 4.049	4.938 3.727	2.109 1.913
1E-5	mean median	1.198 1.072	2.157 1.913	3.044 2.662	2.714 2.491	1.550 1.526	.858 .832
2E-5	mean median	1.071 .973	1.919 1.726	2.696 2.415	2.443 2.266	1.376 1.368	.758 .736
1E-4	mean median	.795 .730	1.402 1.311	1.964 1.804	1.799 1.963	1.007 1.018	.543 .528
2E-4	mean median	.674 .630	1.195 1.129	1.652 1.544	1.542 1.467	.857 .866	.448 .446
1E-3	mean median	.423 .400	.729 .710	1.029 .976	.985 .952	.544 .548	.283 .278
2E-3	mean median	.334 .320	.552 .531	.783 .750	.755 .733	.423 .431	.221 .217
$1.715X10^{-6}$ (2xSSE)	mean median	1.516 1.293	2.813 2.384	3.972 3.280	3.461 3.034	2.060 1.933	1.128 1.081
$1.386X10^{-4}$ (SSE)	mean median	.735 .681	1.301 1.227	1.810 1.676	1.673 1.583	.934 .945	.522 .507

The mean magnitude **M** and distance **R** that cause exceedences of ground motions at specified amplitudes were also investigated to gain an understanding of the characteristics of earthquakes that dominate the seismic hazard. These parameters were calculated for both 10 Hz and 1 Hz motions, and for several amplitudes. Table 3.1-5 shows the calculated values and how they vary with structural frequency and amplitude. The choice of amplitudes was made to obtain results for hazards at approximately $1.4X10^{-4}$ and $1.7x10^{-6}$ annual probability of exceedence.

As indicated by the Table 3.1-5, the mean magnitude **M** increases for higher levels of shaking.

TABLE 3.1-5
MEAN MAGNITUDE VERSUS PSA

Level	Freq.	PSA (g)	$\bar{M}$	$\bar{R}$ (km)
SSE	10 Hz	1.2	6.7	9.3
SSE	1 Hz	1	7.0	17.0
2xSSE	10 Hz	3	6.9	8.7
2xSSE	1 Hz	2	7.2	20.2

Vertical Ground Motions

The attenuation of vertical component ground motion has not been studied as extensively as the horizontal component. As a result, there is only one applicable vertical attenuation relation for spectral values that has been published: Campbell (1990). Unlike the horizontal component, for which 5 different attenuation models were used in the hazard analysis, the vertical attenuation is represented only by the Campbell (1990) model.

Figure 3.1-24 shows the uniform hazard spectra at the SSE level acceleration at 5% damping for the vertical ground motions. The peak spectral acceleration occurs at 10 Hz. The uncertainties represented by the 85th and 15th fractiles are relatively small because only one attenuation equation is used to predict the ground motion. The numerical values for the mean are contained in Table 3.1-6.

The Campbell (1990) model for vertical spectral values has very large standard errors for the high frequency (e.g., 10 Hz) response spectra. This large standard error at 10

Hz has an impact on the high frequency spectral shape for the 2xSSE level. The vertical spectral shapes (normalized over 1-10 Hz) from the hazard study are shown in Figure 3.1-25. There is a significant difference in the spectral shapes between the SSE and 2xSSE levels.

TABLE 3.1-6
MEAN VERTICAL GROUND MOTIONS AT SSE LEVEL PROBABILITIES OF EXCEEDENCE

Frequency	S_a(g)*
0.5 Hz	.247
1 Hz	.439
2.5 Hz	.783
5 Hz	1.312
10 Hz	1.544
25 Hz	.845

* Vertical S_a levels were chosen at an annual probability level of $1.386X10^{-4}$, corresponding to the horizontal SSE.

This difference in the spectral shapes is primarily due to the standard errors in the Campbell (1990) model. To demonstrate this, a simplified analysis was conducted computing the hazard for the SCOZD only. (The SCOZD dominates the high frequency hazard at the SSE and 2xSSE levels at the site.)

An attenuation relation for the vertical component for rock sites was developed by Sadigh et al. (1993). This model presented standard errors that were developed for both soil and rock sites as is commonly done in developing attenuation relations. (For example, the Sadigh (1994) model used for the horizontal component uses standard errors that were developed from a combined set of soil and rock data, but with different median attenuation relations for soil and rock sites.) Therefore, the vertical component standard errors developed by Sadigh et al (1993) are applicable to soil sites as well as rock sites. Since this model is based on a much larger data set than Campbell (1990), the standard errors should be more accurate (and more stable). These standard errors are compared to the Campbell (1990) standard errors in Figure 3.1-26. Preliminary results of new attenuation relations for the vertical component by Abrahamson and Silva (1995) have found standard errors for the vertical component that are similar to the Sadigh et al values.

To test the sensitivity of the spectral shapes at the SSE and 2xSSE levels to the standard errors, the hazard from the OZD was computed using the Campbell (1990) median attenuation relation with two different standard error models. In the first case, the standard errors published by Campbell (1990) are used; in the second case, the Sadigh et al (1993) standard errors for the vertical component are used.

The resulting spectral shapes at the SSE level are shown in Figure 3.1-27 for the full hazard analysis, and for the two simplified analyses. The simplified analyses give similar spectral shapes as the full hazard analysis (Figure 3.1-27) indicating that it is reasonable to use the simplified analysis for this sensitivity study.

The comparison of the spectral shapes at the 2xSSE level is shown in Figure 3.1-28. Using the Sadigh et al standard errors, the spectral shapes for the 2xSSE level is similar to the spectral shape for the SSE level.

Based on this comparison, the vertical spectral shape for the SSE level computed in the hazard analysis is used for both the SSE and the 2xSSE levels.

3.1.4 SEISMIC HAZARD SUMMARY

The seismic hazard results represent the annual frequency of exceedence of various ground motion levels at SONGS, and the uncertainty in the annual frequency of exceedence. These results are represented as a family of fractile seismic hazard curves, and as uniform-hazard spectra corresponding approximately to the SSE and 2xSSE levels. The uncertainties in hazard are derived from uncertainties of input assumptions regarding seismic sources, seismicity parameters, and ground motion attenuation equations. Thus, the analysis performed for SONGS is state-of-the-art, because it incorporates and presents uncertainties in the major factors affecting seismic hazard in the region around the site.

The tectonic interpretations and seismicity parameters were derived by Geomatrix Consultants, Inc. They consist of faults and area sources in southern California that might contribute to hazard, and the parameters defining those sources (slip rate, *b*-values, geometry, and maximum magnitude). For area sources, an analysis of historical seismicity was conducted to establish rates of activity and *b*-values.

Attenuation equations were derived by Woodward-Clyde Consultants. Five equations were selected to estimate spectral velocity at 25, 10, 5, 2.5, 1, and 0.5 Hz. These were weighted by comparing the predictions to observations of strong motion in southern California and determining the goodness of fit. All five equations predict ground motion on stiff soil, which is appropriate for application at SONGS.

The methodology in this study follows closely that used in other state-of-the-art studies of seismic hazards at nuclear plant sites in the U.S. The derivation of seismic sources is specified by the earth science experts; an analysis of historical seismicity is performed to aid in estimation of seismicity parameters; and all relevant theories and data on earthquake causes and characteristics in southern California were examined and incorporated into the interpretations.

This study used three teams to develop the seismic hazard results, one for attenuation equation, one for seismic source descriptions, and one for integration and hazard calculations. In addition, an expert review panel was assembled and has provided feedback during the entire project, including reviews of intermediate results and reports and the final report. Thus, the results presented here have a strong basis and are appropriate for use in the IPEEE PRA for SONGS. Regarding the analysis of earthquake data, an extensive evaluation of the earthquake catalogs used in this study and described in Section 3.1.2.2 was not conducted to address issues such as the accuracy of specific event locations and magnitudes, the conversion of intensity to magnitude, and the completion of earthquake coverage represented by the catalogs. The available catalogs have been scrutinized closely (e.g., by Ellsworth of the USGS) and it is appropriate to use these data bases as presented. The catalogs have simply been accepted and used with their listed values of magnitude and location. Similarly, the specific soil conditions at SONGS have not been modeled in detail with the ground motion attenuation equations adopted here. The attenuation equations use generic factors to estimate the dynamic response of stiff soils. Site-specific studies of soil response under earthquake loads might yield results different from those used here, with a corresponding effect on the hazard results. Also as pointed out regarding the comparison of predictions and data in Reference [3-1], there is an apparent tendency for the attenuation equations to slightly over-predict spectral response, which if correct would result in the hazard values reported here being slightly conservative.

3.2 PLANT INFORMATION AND SELECTION OF SYSTEMS AND EQUIPMENT

3.2.1 METHODOLOGY AND REVIEW PROCESS

This section describes the selection of seismic events, systems, structures, components and relays used in the seismic IPEEE. The evaluation of seismic relay chatter is also described.

Selection of Seismically-Induced Events

The strong ground motion of a seismic event may challenge the plant and cause a secondary event that disrupts normal plant operating conditions. These secondary seismically induced events include initiating events identified in the IPE (such as uncomplicated reactor trip, loss of offsite power, LOCA, steam generator tube rupture, steam line break, etc.). These secondary events also include seismic-induced fires and floods. Seismic-induced fires and floods are addressed in Section 3.3.4.

Based on seismic fragility walkdowns and/or seismic capacity calculations, most of the IPE initiating events were screened based on high capacity with very low likelihood of failure. Events and components were screened from further analysis if the likelihood of the seismic-induced event or seismic-induced failure of a component is less than 1E-7/year. To assess the approximate fragility which corresponds to the screening criteria, EQESRA (seismic quantification code) was run with the SONGS 2/3 seismic hazard curve. After several sensitivity runs, it was determined that the screening fragility was approximately 8 gSA depending on the uncertainty factors (β_R, β_U). Components with fragilities in the range of 7-10 gSA were individually evaluated to determine if seismic-induced failure is less than 1E-7/year. Table 3.2-1 lists the IPE initiating events and their potential to be seismically induced.

Events such as uncomplicated reactor trip, loss of offsite power and small LOCA, however, could not be screened. All front-line and support systems necessary to mitigate the impact of these remaining seismic-induced events were modeled in the seismic event tree. The purpose of the seismic event tree is to identify and quantify sequences where seismic failures lead to either 1) core damage or 2) degraded plant states. The degraded plant state sequences are transferred to the appropriate IPE event trees to identify and quantify additional non-seismic random failures which would lead to core damage. The SONGS 2/3 seismic event tree is detailed in Figure 3.6-2.

Containment bypass events were also examined. Potential bypass sequences such as interfacing systems LOCA, catastrophic failure of the reactor pressure vessel, and steam generator were also considered. Based on seismic fragility walkdowns, these sequences have been screened based on high component capacities.

TABLE 3.2-1
IPE INITIATING EVENTS INCORPORATED INTO THE IPEEE

IPE Initiating Events	Disposition
Turbine Trip	Assumed to occur given a loss of offsite power.
Loss of Power Conversion System (MFW or Condensate System)	Assumed to occur given a loss of offsite power.
ATWS	**Included in IPEEE**
Loss of Offsite Power/SBO	**Included in IPEEE**
MSLB/FLB	Fragility assessment verified that MFW/AFW & steam lines are seismically rugged. Therefore, these events are screened from further analysis.
Medium and Large LOCA	Fragility assessment verified that reactor coolant system piping and associated RCS components are seismically rugged. Therefore, these events are screened from further analysis.
Small & Small-Small LOCA	**Include in IPEEE**
SGTR	Fragility analysis verified that the SG and internals are seismically rugged. Therefore, SGTR is screened from further analysis.
Interfacing Systems LOCA (including SG, RCS failure)	Fragility assessment verified that low to high pressure systems interacting with the RCS are seismically rugged. Therefore, ISLOCA is screened from further analysis.
Reactor Pressure Vessel Rupture	Fragility assessment verified that the reactor vessel is seismically rugged. Therefore, vessel rupture is screened from further analysis.

Selection of Seismic Equipment

To assess the seismic fragility of the mitigating safety systems, the associated components and structures were identified and placed on the seismic equipment list (SEL). The SEL is provided in Section 3.11.1. Additional guidelines [3-24] used to develop the SEL are listed below:

1. All front line safety systems and support systems are included in the SEL. Buildings and structures are evaluated with the system.

2. Check valves and manual valves are generically rugged and are not included in the SEL.

3. Breakers, cable trays, circuit boards are generically rugged and are not included in the SEL.

4. Piping is generically rugged and is not included in the SEL.

Based on these guidelines, the SEL lists approximately 600 pieces of equipment. The SEL includes but is not limited to:

- pumps
- valves
- tanks
- chillers
- accumulators
- relay and control cabinets
- spray headers
- control panels
- transmitters
- motor control centers
- transformers
- transfer switches
- process switches
- inverters
- switchyard
- batteries
- battery chargers
- heat exchangers
- diesel generators and auxiliaries
- control room
- alarms
- RCS & internals
- all containment integrity components
- pressure regulators

To assess whether these components must be added to the plant seismic model, the seismic fragility for each of the SEL components was assessed. The seismic equipment list and fragility analysis results are provided in Section 3.11.1. The fragility analysis is further described in Section 3.4.

The Nuclear Safety Group developed the SEL and the Nuclear Engineering Design Organization independently reviewed the SEL for completeness, comprehensiveness and accuracy.

Selection of Electrical Relays and Relay Chatter Analysis Methodology

Past seismic PRA studies have shown that seismic-induced relay chattering may have a significant impact on the availability of systems to properly function. Among many possible effects, relay chattering may result in inadvertent operation of equipment, rapid cycling of equipment, or prevent equipment from operating when required. To assess the impact of relay chatter on SONGS Units 2/3, electrical relays for each component on the SEL were identified and placed on a seismic relay list (SRL). Relays determined to be seismically rugged were screened from further analysis. Relay chatter fragility assessments are further discussed in Section 3.5.

Relay Chatter Evaluation Methodology

The relay chatter evaluations for each of the relays were performed consistent with EPRI-NP-7148-SL [3-21]. Based on the chatter evaluation, each relay was identified as either chatter acceptable (CA), chatter unacceptable (CU), or operator action (OA) (where an operator is required to mitigate chatter impact). Those relays that are CU or OA were modeled in the seismic event tree nodal equations. The operator actions for the OA relays were evaluated and included with the OA relay in the seismic event tree nodal equations.

To assess whether a relay is CA, CU, or OA, the elementary diagram for each relay was examined. For each relay chatter evaluation, guidelines provided by EPRI-NP-7148-SL were used. In addition, the following guidelines were used:

- Multiple relay chatters within a circuit must be considered unless contacts are assessed to be seismically rugged.
- The duration of strong seismic ground motion which may result in relay chatter is approximately 20 seconds. Strong motion is defined as motion that is structurally significant which for SONGS is peak ground acceleration excursions on the order of 0.5g or greater. The strong motion will be typically about 10 to 15 seconds. Therefore, 20 seconds will be a conservative estimation of the strong motion for the SONGS IPEEE.
- Based on fragility evaluations, seismic walkdowns, and accident analyses, the following ESFAS signals would not be expected to actuate during the period of strong ground motion: Safety Injection Actuation Signal (SIAS), Containment Isolation Actuation Signal (CIAS), Containment Spray Actuation Signal (CSAS), Recirculation Actuation Signal (RAS), and Main Steam Isolation Signal (MSIS).

- An Emergency Feedwater Actuation Signal (EFAS) may be generated during the 20 seconds of strong ground motion. Relay chatter is evaluated with both EFAS actuated and not actuated.

- A Loss of Voltage Signal is assumed to be generated following a loss of offsite power.

- Based on EPRI NP-7148-SL, the following types of mechanical or solid state devices shown in the logic signal strings of Elementary Diagrams (E/Ds) are seismically rugged and therefore do not chatter:

  a. "42" - Relay Contacts (Mechanical device - Running Circuit Breakers)
  b. Limit Switch Contacts: Torque and Position
  c. "33" - Relay Contacts (Position Switches)
  d. "TS" - Test Switch Contacts
  e. "CS" - Control Switch Contacts
  f. "HS" - Hand Switch Contacts
  g. Solid State Devices

The seismic relay list consists of over 1300 relays of which 191 relays[1] are modeled as chatter unacceptable, and 27 relays require operator actions to mitigate the effects of relay chatter. The remaining relays are either seismically screened due to ruggedness or whose chatter have acceptable consequences. Section 3.5 provides a discussion on the fragility evaluation of the SEL relays.

The Nuclear Safety Group developed the SRL and completed the relay chatter evaluations. The Nuclear Engineering Design Organization's Electrical Group independently reviewed the evaluations for completeness, comprehensiveness and accuracy.

3.3 WALKDOWNS

In accordance with Generic Letter 88-20, Supplement 4 [3-22], and NUREG-1407 [3-23], a documented walkdown of the items included on the Seismic Equipment List (SEL) was performed for the purpose of identifying equipment/system seismic vulnerabilities in either the component load path or anchorage, potential seismic failure/falling and proximity interactions, and potential flooding or fluid spray interactions, including multiple concurrent flooding sources when credible. The basis, scope and results of the

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[1] 171 of the 191 relays are switchyard relays. In the plant model, the switchyard relays have been grouped as one switchyard relay with a conservative fragility selected.

essential equipment walkdowns are described in this section and include the summary of the walkdown effort, and the major observations and findings.

The objectives of the seismic walkdown were to:

1. Review and gather detailed information and measurements on equipment and structures, search for potential seismic vulnerabilities, as well as review potential spatial system interaction concerns.

2. Add to the Screening and Evaluation Worksheets (SEWS) any items which the team judges to be potentially serious problems (such as air lines to air-operated equipment, questionably secured space heaters, lights over critical batteries, etc.).

3. Evaluate the seismic ruggedness of tanks, vessels, and pipes that carry flammable materials. Fire protection systems in the plant will be reviewed for seismic-induced fire and inadvertent actuation of fire protection systems. Seismic failures of cardox/Halon systems as well as water suppression systems will be evaluated.

4. Note questionable seismic practices as concerns (poor housekeeping, non-tied-back gas bottles, heavy unanchored equipment, deficient seismically-supported water or fuel storage tanks, or seismically vulnerable piping).

5. Review generally rugged components (such as piping, cable trays, conduits, check and manual valves) for potential seismic interaction with other equipment and structures.

The observations and findings of the walkdowns are given in Section 3.3.3. Calculations were performed for components that could not be screened on the basis of the walkdowns.

Walkdowns were conducted by seismic capability engineers and systems analysts to cover the scope of the seismic IPEEE. The seismic capability engineers have completed the SQUG Walkdown Screening and Seismic Evaluation Training Course plus the add-on Seismic IPE Course, and have performed several seismic walkdowns.

Most items on the SEL were found to be well engineered, well anchored and seismically rugged. Equipment seismic qualification analyses and anchorage calculations were found to be available for most components. Some items which were not screened on the basis of the walkdowns were shown to have adequate margins by subsequent calculations. The remaining unscreened items were included in the seismic model.

3.3.1 METHODOLOGY AND REVIEW PROCESS

Several walkdowns were conducted to complete the walkdown effort. The Unit 2 walkdown was primarily conducted by SCE engineers and the Unit 3 walkdown was conducted by EQE engineers. Joint walkdowns by SCE and EQE personnel were conducted to establish consistency in the resulting inspections and evaluations, as well as to review each others judgements and to establish the similarity between the two units. A preliminary walkdown to establish the preferred and alternate shutdown paths for seismic IPEEE was performed by the following individuals:

1. G. Hardy, EQE, Seismic Capability
2. T. Kipp, EQE, Seismic Capability
3. J. Appel, SCE, Civil Engineer
4. D. Ostrom, SCE, Seismic Capability
5. T. Yee, SCE, Seismic Capability

Subsequent walkdowns were conducted in 1993 to 1995. The walkdown teams were formed using the following individuals :

Unit 2	Unit 3
1. D. Ostrom, SCE, Seismic Capability	1. G. Hardy, EQE, Seismic Capability
2. J. Appel, SCE, Civil Engineer	2. T. Kipp, EQE, Seismic Capability
3. T. Yee, SCE, Seismic Capability	3. T. Roche, EQE, Seismic Capability
4. R. Blaschke, SCE, Seismic Capability	4. M. Salmon, EQE, Seismic Capability
	5. J. Appel, SCE, Civil Engineer

The final Seismic Equipment List (SEL) is given in Section 3.11. A preliminary SEL was used in the initial walkdowns. The final database evolved through discussions between SCE and EQE and was finalized during the walkdowns. All listed components were visually evaluated except for a few components which were assessed to be radiologically inaccessible. The fragility assessment for these items is based on component and anchorage drawings.

The finalized SEL was used as the basis for the planned and documented seismic capability walkdowns. Some additional walkdowns were performed later to encompass equipment added to the list and to obtain additional data for the fragility calculations.

The SEWS, in conformance with EPRI NP6041-SL, were either programmed into a Convertible Grid Computer or manually filled-out [3-24]. They are identical to the SEWS established in Appendix F of Reference 3-24 which provides the technical guidance for the IPEEE walkdown effort.

Formal documentation was prepared for all major equipment items and for any usually robust components which exhibited potential seismic concerns. As an example, if a motor operated valve was evaluated which satisfied all the seismic caveats (e.g., operator weight and eccentricity were within limits, valve and yoke did not include cast iron materials, the valve and operator were not independently supported), the valve was screened out and not identified as a potential outlier. However, if the operator for that same valve was in close proximity to a structural steel member, the valve was identified as an outlier because of the potential seismic interaction and formal documentation prepared describing the condition.

When the team had a reasonable basis for assuming that a group of components were similar and were similarly anchored, then it was only necessary to thoroughly inspect one component out of this group. For example, only one of the four containment fan cooler units required a thorough inspection. For the other three fan coolers, the "walk by" established the principal of "similarity" and the absence of any system or spatial interaction (such as a block wall). Actual similarity was verified during the "walk by." Traceability to the one inspected item and the basis for similarity were recorded on the SEWS.

3.3.2 EQUIPMENT CAPACITY WALKDOWNS

Most items are robust. The SEWS note whether or not the items could be screened as a result of the walkdown observations coupled with the walkdown team knowledge of the design basis conservatism at the time of the walkdowns. Subsequent screening calculations and review of qualification calculations resulted in adequate fragilities for most of the items that were not screened on the basis of the walkdowns.

Anomalies observed in the walkdowns are identified in the SEWS. These are denoted with a "No" answer in the "Caveats Satisfied?" column and indicate that one or more caveats in the SEWS forms was not met. The notable anomalies observed during the walkdowns are listed below with resolution noted in the parentheses. Some of the anomalies were resolved when the structural response analyses were completed and differential displacements were determined to be relatively small. Other anomalies were resolved upon completing seismic fragility calculations or determining the need for corrective actions as noted in Section 3.3.6.

3.3.2.1 Issues Potentially Affecting Functionality

1. Remote valve function control may be affected by the orientation of hand wheel key during a seismic event. (Occurrence was judged to have a very low probability.)

2. CCW surge tank level controller is a magnetrol-type which may function poorly in seismic events. (Tank level instrument will not adversely affect system function.)

3. Turbine governor valve is a specialized valve with linkages and springs. This valve is outside of normal valve configuration. (Fragility calculation performed.)

3.3.2.2 Anchorage Anomalies

1. Cantilevered friction blocks are employed to restrain Reserve Auxiliary Transformers. (Transformer has a low fragility.)

2. Spacing violations on CCW heat exchanger anchorage. (Fragility calculation performed.)

3. Anchorage on transfer switches for HPSI and CCW were not accessible during walkdown. (Drawings used to confirm anchorage details.)

3.3.2.3 Load Path Anomalies

1. Potentially large nozzle loads due to limited support of discharge piping over a relatively long span on HPSI and LPSI pumps. (Fragility calculation performed.)

2. Poor weld conditions between the cylindrical vessel and the support legs for several accumulators, including 1/8" gap and non-standard stitch welding. Burn through observed on legs. (Fragility calculation performed.)

3. No bolt chairs seen on the non-safety-related condensate storage tank because the anchorage was covered. The shell/base flange connection may be the weak link. (Fragility calculation performed.)

4. Potentially heavy hydraulic operator set on top of valve and unsupported laterally. (Valve and valve operator are adequately supported by piping.)

5. Potentially inadequate mounting of internal devices which are offset from the back of the panel on emergency chiller unit. Vibration isolators are placed between the panel and support brackets and offset from the internals. Potentially ineffective hinged door panel clips. (Fragility calculation performed.)

6. Axial restraint for emergency chiller compression tank is dependent on friction. Bolts may not be able to maintain tightness. (Fragility calculation performed.)

7. 3/4" RCP bleed to VCT piping is supported from containment penetration while the motor operator support is attached to the interior concrete structure. (Relative displacements are small.)

8. Lateral restraint of the coil on the diesel engine radiator fan is not evident. It appears to be guided to allow for thermal expansion. (Fragility calculation performed.)

9. Marginal attachment of motor oil filter for emergency chiller unit. (Rechecked and concluded to be acceptable.)

3.3.2.4 Seismic Interaction II/I Issues

1. Overhead lighting fixture hung from the typical hardware by bailing wire. (Temporary condition during refueling outage.)

2. Trickle charger sitting on a wall bracket is potentially a falling source for battery. (Charger restrained by two bolts connected to bracket.)

3. Unrestrained floor-mounted fire extinguisher within 12" of stainless steel tubing between accumulator and valve. (Temporary condition during refueling outage.)

4. Flexible conduit system with limited support in the vertical and transverse directions on diesel fuel transfer pumps. In addition, unencased instrumentation wiring of roughly 4 feet. (Supports judged to be adequate.)

5. One end of light fixture broken away from unistrut support which employs generic rod support on toxic gas monitoring panel. (Repaired.)

6. Rigid support point on the incoming chilled water line on CCW pump room emergency AC unit. (Relative displacement is small.)

7. Instrument line from head of compression tank rigidly restrained in two directions and is a possible hard spot if the frame is flexible. (Relative displacement is small.)

8. Possible break of large sight-glass on the RCP oil drain collection tank due to potential tank distortion or impact by falling items. (Reverified that there are no potential falling objects.)

9. Control room ceiling was inaccessible. (Ceiling was checked in a subsequent walkdown.) Unanchored bookshelves and unrestrained equipment behind the Technical Support Center windows are potential interaction hazards. (Bullet-proof glass prevents items from falling into the control room below.)

3.3.2.5 Commodity Clearance Issues

1. Possible impact of unanchored supply cabinet with accumulator may cause failure of relief valve threaded connection. (Temporary cabinet during refueling outage.)

2. AFW valve operator confined by floor grating with 0" clearance to motor and housings. May not be adequate for relative displacements. (To be repaired as scheduled in Section 7.1.)

3. 1/16" clearance between the control panel and a 1" tube running beside it could affect the operability of relays on emergency chiller units. (Relative displacements are small.)

4. Limited clearance between the containment sump valve to radwaste isolation valve (MOV) operator and an adjacent piping support attached to the interior concrete structure. (Relative displacements are small.)

5. 0" clearance between various unbolted cabinets. (Cabinets containing essential relays with unacceptable relay chattering consequences to be repaired as scheduled in Section 7.1.)

6. Limited piping flexibility to accommodate relative structure displacements. (Relative displacements are small.)

3.3.3 RELAY WALKDOWN

As stated in EPRI NP-7148-SL [3-32], the purpose of the relay walkdown was to:

- Obtain information needed to determine cabinet types which house essential relays and to determine their dynamic characteristics and in-cabinet amplification for the seismic capacity screening.

- Spot check relay mountings.

- Spot check relays types and locations, including checks for vulnerable relays.

- Verify the adequacy of the anchorage of the cabinets/enclosures which support the essential relays.

The SONGS 2/3 relay walkdown was generally performed in conjunction with the equipment capacity walkdowns. The primary objective of the relay portion of the walkdowns was to verify that the relays were mounted in a sound manner. To meet these objectives, at least one cabinet of each type and each train was evaluated by looking at the relays and mountings within all the electrical cabinets, panels, and switchgear. In addition relay mounting was spot-checked for each cabinet and panel that was opened for inspection. This activity was performed by the seismic capability engineers. As needed, information was obtained on the cabinet/enclosure type, anchorage, internal panel cut-outs, and relay mounting.

In addition, other walkdowns were performed in order to determine the type and model number for some relays, or their position in a cabinet (for potential seismic-induced floods). Since some relays were determined to be replacements, the replacement procedures at SONGS 2/3 were verified to include an evaluation of the seismic qualification of the replacement relays.

There were no problems of loose mounting or relays mounted on panels with excessive cut-outs or flexibility. However, the hinged interior doors on one set of cabinets were found to have missing nuts for the bolts to the cabinet frame, which could have led to excessive vibration. A maintenance action was requested and confirmed to have been performed.

This overall review of relays and mountings provides high assurance that the essential relays are properly installed.

3.3.4 SEISMIC FIRE/FLOOD INTERACTIONS AND WALKDOWN

This section documents the evaluation of potential seismic-fire and seismic-flood interactions. Additional insights on the effects of inadvertent fire protection system actuation, listed in NRC Information Notice 94-12, were also included in this evaluation process [3-25, 3-29]. The purpose of this evaluation was to:

- identify any potential for a seismic-induced fire or seismic-induced flood that could damage equipment or structures that are important to safety during or after a seismic event, and

- determine whether any identified damaging seismic-induced fire or floods are vulnerabilities and to assess viable mitigating measures.

3.3.4.1 Background

This section is divided into two main sections corresponding to seismic-induced fire and seismic-induced flood. Seismic-induced inadvertent actuation of fire suppression systems, including Cardox and Halon, is considered under the seismic-induced flood section. The general approach and detailed results, as documented in the Seismic-Fire and Seismic-Flood Evaluation Project Instruction [3-26] and associated analysis file [3-27] is described below:

Step 1. Identification of potential sources of fire or floods. All potential sources of fires or floods that could damage safety equipment or structures (that is, equipment that are included in the seismic equipment list or the structures that contain this equipment) were identified. These sources were either identified from existing documents or identified during the walkdowns.

Step 2. Evaluation of seismic capacity of sources. The seismic capacity of the identified fire or flood sources was assessed. Many potential fire or flood sources may have been screened out using expert judgment during the seismic capacity walkdown based on high seismic capacity. However, in some cases it may have been necessary to formally calculate the seismic capacity of the fire/flood source.

Step 3. Assess impact of fire/flood source on core damage frequency. If the seismic capacity value was less than the capacity screening level, then the potential for fire or flood damage was evaluated by:

- Inclusion of the fire/flood source in the seismic systems model and quantified to determine its impact on seismic core damage frequency

- Development of a sensitivity study to evaluate the potential for adverse impacts on seismic core damage frequency

The capacity screening level is defined by the project instruction as the average spectral ground acceleration associated with a 1E-6/year seismic return frequency. A more conservative screening criterion of 1E-7/year was actually used for the evaluation, which corresponds to a median capacity of 7g to 8g, depending on the uncertainty parameters. This was consistent with the balance of the seismic capacity analysis.

If the fire/flood source was determined to be a significant contributor to seismic core damage frequency, then it was evaluated to determine if it is a seismic vulnerability and if modifications would be recommended.

3.3.4.2 Seismic-induced Fire

For seismic-induced fires, the concern was that the earthquake motion could cause a failure of a tank or piping containing flammable liquids or gases[2], which are then ignited causing a damaging fire. The primary fire sources are transformers, tanks and piping containing pressurized hydrogen or flammable gas, or oil/fuel (such as hydraulic oil, cooling oil, fuel oil, lube oil, and waste oil). Simultaneous random fires (i.e., not caused by the seismic event) were not evaluated, since the combined probability of a major earthquake and an independent random fire which damages equipment important to seismic safety is very small. For example, cable tray or trash barrel fires are generally not caused by an earthquake and therefore were not evaluated in the seismic fire evaluation. While there may be some potential for a seismic-induced cabinet fire due to cable shorting at the entry to the cabinet, the essential electrical cabinets and associated top-entry cables were evaluated for their seismic capacity as part of the overall capacity evaluation of electrical equipment. Seismic capacity evaluations for

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2 For this evaluation, "flammable" is defined to include "flammable" and "combustible".

essential pumps included a capacity evaluation of the lube oil system and lube oil reservoir. Similarly, the capacity evaluations for the diesel generators and day tanks included assurance of confinement integrity for flammable liquids.

Step 1. Identification of the Potential Fire Sources

The Updated Fire Hazard Analysis (UFHA) and associated Appendix R documents identified sources and quantities of flammable liquids and gases by fire areas. Table 3.3-1 lists the potential sources included in the walkdown.

Although welded piping generally has very high seismic capacity, a special effort was made to identify significant piping containing flammable liquids and gases in order to ensure that the flammables in these pipes do not pose a seismic-fire interaction. In particular, piping containing hydrogen, waste gases, and other flammables were identified and reviewed as follows:

- Most of the waste gas lines are in the radwaste areas. The radwaste areas do not contain components that are important for safety during or after a seismic event, except for the charging pump rooms (which are isolated from the rest of the radwaste areas). These areas were therefore screened from further analysis. Excluding the radwaste area, the areas containing lines with the potential for hydrogen leakage are listed in Table 3.3-2. In general, the quantity of flammable gas in these low pressure lines is insufficient to pose a serious fire hazard. However, the areas were listed for seismic capacity walkdowns.

- Hydrogen leakage from potential piping failures could also occur in the containment, turbine building, and yard areas. Hydrogen supply piping running from the storage tanks near the turbine buildings to the radwaste area along the outside of the control and penetration areas was identified. While potential leakage and fire in outside areas would not damage any safety-related equipment, these pipes were listed for walkdown to ensure that there was no potential for piping failure or fire propagation. Based on containment walkdowns and the piping walkdowns, it was judged that the waste gas and sampling lines inside containment are very rugged. There are no pure hydrogen lines inside containment, and the amount of flammable gas in the waste gas and sample lines inside containment is relatively small. Based on the above considerations, and for ALARA reasons, no additional containment walkdowns were performed for the seismic-fire interaction issue.

- Piping containing flammable liquids, such as the diesel fuel oil lines were not listed separately for walkdown, but were examined during both the seismic-induced fire walkdown and the equipment seismic capacity walkdowns.

Because of their temporary nature, transient liquid and gas fire sources were not categorically identified for walkdown. However, the walkdown team did identify and review potential transient combustibles that were found during the walkdown of other fire sources and areas.

The results of this step are the lists of potential fire sources and quantities for the seismic-induced fire walkdown (Table 3.3-1), and potential piping and tanks with hydrogen (Table 3.3-2).

Step 2. Performance of the Seismic-induced Fire Walkdown to Screen Fire Sources

A walkdown was performed to evaluate the seismic capacity of these potential fire sources. The walkdown team included seismic capability engineers, fire protection engineers, and PRA systems engineers. Each area containing potential fire sources and safety equipment was walked down. The fire protection engineers identified the potential fire sources, and the seismic capability engineers evaluated each source by checking the anchorage and capacity of the fire source (e.g., tank) and attached piping, and the potential for nearby spatial interactions. Screening and Evaluation Worksheet (SEWS) forms were available to record seismic capacity information, but were not needed for this evaluation since none of the potential fire sources were judged to be a problem.

The walkdown results show that all potential seismic fire sources either have high capacity and could be screened from further analysis, or that there are no safety equipment or cabling in the vicinity of the potential fire impact area. For example, the waste lube oil tanks for the reactor coolant pumps were evaluated to have high seismic capacity and will not fail during a seismic event. Also, the hydrogen storage tanks were screened from further analysis because they are located on the side of the turbine building that is away from containment, where their potential failure could not damage safety equipment. The walkdown results are documented in Table 3.3-1.

The walkdown team paid particular attention to piping containing hydrogen and other flammable gases. The hydrogen supply piping was traced both inside and outside the plant buildings. These lines were spot-checked and found to consist of welded piping that is well supported, and are not located near any safety equipment. Waste gas lines which could contain hydrogen and are in areas with safety equipment were also spot-checked. These lines are also welded and well-supported, and would have high capacity during an earthquake.

In summary, there were no potential failures identified in hydrogen or waste gas piping which could impact safety equipment. These walkdown results are documented in Table 3.3-2.

Step 3. Seismic-induced Fire Results

Based on the systematic identification and evaluation of fire sources, no fragility calculations were required, and potential seismic-induced fires were screened from the seismic analysis.

3.3.4.3 Seismic-induced Flood

The concerns associated with seismic-induced flood can be categorized as:

1. Seismic failure of tanks, piping, expansion joints, and seals that cause flooding or spraying that damages equipment important to seismic safety.

2. Seismic-induced inadvertent actuation of fire suppression systems causing flooding, spraying, or Cardox/Halon discharge damaging seismic safety equipment.

Note that some equipment may not be susceptible to damage even if sprayed. For example, many valves are designed to withstand spraying and operate as designed. Also, discharge of Cardox/Halon due to piping failure will not damage electrical or mechanical equipment, or cables. As with seismic-induced fires, the general approach included identification of potential seismic-induced flood sources, and performance of a seismic capacity walkdown of the sources. The potential for failure of multiple flood sources was included in the assessment when considered credible. The walkdown team included seismic capability engineers to judge flood source seismic capacity on a screening basis, fire protection engineers to provide information on fire suppression systems and flood impacts, and PRA engineers to provide systems and IPE expertise. The steps used to evaluate these concerns, and the walkdown results are discussed in the following sections.

Step 1. Identification of Potential Flooding Sources

The sources of potential flooding, including piping, tanks (inside and outside of buildings), fire sprinkler and suppression systems, and circulating water system condenser expansion joints and seals, were identified from the internal flooding analysis in the IPE, and the analysis of inadvertent fire suppression system actuation performed for the Appendix R evaluation.

Step 2. Seismic Walkdown and Evaluation of Potential Flooding Sources

The walkdown and evaluation of each of the potential flood source categories are discussed below:

A. Piping (Including Fire Water Systems)

In general, piping has very high seismic capacity. Potential piping failures at SONGS have been examined previously during seismic II/I evaluations. For the IPEEE, potential piping failures near safe shutdown equipment were reviewed during the seismic capacity walkdown as part of the systems interaction evaluation and documented on SEWS form. In particular, threaded piping (such as fire sprinkler systems in the vicinity of equipment on the seismic equipment list) was verified to have adequate support and seismic capacity such that failure will not cause flooding or spraying. Some sprinkler or fire suppression systems were identified as "dry pipe" systems where piping failure would not directly cause discharge of water or Cardox/Halon.

Fusible links in sprinkler systems generically have high seismic capacity and can be screened out unless subject to unusual stresses such as impact with adjacent objects/structures. Potential interactions of these sprinkler heads were reviewed during the walkdowns with no issues or weaknesses identified.

B. Tanks

The internal events IPE flooding analysis was reviewed to identify tanks that could fail and cause damage to seismic safety components. This review also included outside tanks that could fail and flood buildings from the outside. These tanks were included in the flood source list as potential flooding sources (Table 3.3-3) and reviewed by the seismic capacity engineers during the walkdowns. In addition, the walkdown team performed a spot-check for any additional tanks in the vicinity of safety equipment that could cause flooding. Several outside yard tanks were added to the list.

The seismic-induced flood walkdown was performed in conjunction with the seismic-induced fire walkdown, and the results are documented in Table 3.3-3. All of the potential flooding sources and piping were determined to either have high seismic capacity, or have no impact on safety equipment given tank failure. However, during the walkdown, the ammonia tank (2/3T-257) was identified as a potential source of toxic gas if failure occurred during a seismic event. Rather than include its potential failure in the seismic analysis, it was decided to strengthen the tank and supports such that the tank would be screened from further analysis. Piping and nozzles were judged to be able to withstand the seismic event due to adequate piping supports and inherent ruggedness. Seismic strengthening of the tank will be completed by the end of the Cycle 9 refueling outage.

C. Expansion Joints And Seals

The seismic capability engineers, as part of the seismic walkdown, evaluated the potential failure of expansion joints and seals.

Two issues were specifically evaluated and documented. First, the seismic capacity of the circulating water system condenser expansion joints was evaluated since failure may cause flooding and failure of safety equipment (including SWC, HPSI, CCW, emergency chillers). However, if offsite power is concurrently lost due to the seismic event, then the circulating water pumps will lose power and a significant flood cannot occur.

A special sensitivity study was performed that demonstrated that the seismic capacity of the expansion joints (about 2.8g S_a) was significantly greater than the capacity of offsite power (about .52g S_a). These seismic fragilities were combined with the seismic hazard curve, and the calculated frequency of expansion joint failure and offsite power *availability* success was determined to be less than 1E-7/year, which is lower than the screening criterion for seismic sequences. Based on this sensitivity study, the potential for seismic-induced expansion joint failure and a significant flood was screened from further evaluation. It should also be noted that additional failures besides the expansion joint failures would be needed to cause core damage, so this sensitivity study is conservative.

The second issue relates to potential seismic damage to seals and bellows around piping into the valve rooms associated with recirculation and to CCW. Failure of seals and bellows allows groundwater to potentially damage required safety equipment. The CCW room seals and the recirculation sump seals were not screened and are included in the seismic model as failure modes for the CCW and Injection/Recirculation systems, respectively.

D. Seismic-induced Inadvertent Actuation of Fire Suppression Systems

In addition to the potential for seismic-induced mechanical failures, potential also exists for inadvertent actuation of fire suppression systems due to seismic-induced phenomena such as relay chatter, dust induced smoke detector actuation, or spurious deluge valve operation. In many cases, inadvertent actuation of fire suppression systems will not damage equipment important to seismic safety. For example, the turbine-driven and motor driven AFW pumps and associated cabinets and valves that are encircled by the deluge system are designed and protected such that they will operate even if the deluge system operates.

NRC Information Notices 83-41, "Actuation of Fire Suppression Systems Causing Inoperability of Safety-Related Equipment" [3-28], and 94-12, "Insights Gained From Resolving Generic Issue 57: Effects of Fire Protection System Actuation on Safety-related Equipment" [3-25, 3-29], expressed several concerns regarding the inadvertent actuation of suppression systems causing damage to equipment credited for safe shutdown. References 3-30 and 3-31 responded to the NRC IN 83-41 concerns, which include some of the same concerns as NRC IN 94-12. The relevant conclusions of the NRC IN 83-41 analysis are as follows:

1. Contamination of diesel fuel oil by fire suppression system water will not occur at SONGS 2 & 3, as the fire suppression system is not connected to the diesel fuel oil tank.

2. Actuation of the water suppression systems at SONGS 2 & 3 due to inadvertent actuation of smoke detectors will not occur, as the alarm and actuation systems are separate. Smoke detectors are used only for alarm. Heat detectors, which are seismically rugged, are used for actuation of the suppression systems.

3. Most plant areas are provided with floor drains sized to remove expected fire fighting water. The analysis performed for NRC IN 83-41 regarding water accumulation in plant areas without drains sized for fire protection systems demonstrated that safe shutdown capability will not be adversely impacted by flooding [3-31]. All areas of the plant containing both automated water suppression systems and safety related equipment were considered in this analysis. The flooding analysis was performed assuming inadvertent operation of the water suppression system for 30 minutes, and flood heights were calculated from known water discharge rates and floor areas. Based on this IN 83-41 analysis, several modifications such as curbs and weatherstripping were made so that

inadvertent operation of the water suppression system would not impact the safe shutdown capability of the plant.

However, the *design basis* flooding analysis did not consider multiple actuations of fire systems, as could occur for seismic-induced relay chatter or spurious deluge valve operation. The only areas which could be affected by multiple actuations of fire systems are those on the nine foot elevation of the auxiliary building. These areas were evaluated during the seismic/flood walkdown by examining the following issues.

- Multiple spurious deluge valve operation due to the seismic event

  While the deluge valves have been tested and are relatively rugged, there have been a few instances when an inadvertent discharge has occurred due to physical impacts to the valve actuation panel. Root cause analysis pointed to improper resetting of the manual pull rod, partly due to corrosion of the valve actuation internals. Immediate corrective actions were taken to identify and correct valves subject to this corrosion, and perform preventive maintenance to prevent the potential for spurious actuation due to impacts.

- Multiple deluge valve operation due to relay chatter

  While the relays associated with the fire protection actuation system are judged to be rugged based on their similarity to known rugged relays, there is no direct capacity information available. Without specific test results, multiple relay chatter could not be excluded.

The walkdown identified potential flood propagation paths and equipment that could be affected by a multiple deluge valve flood. A qualitative analysis was performed to assess the impact and is described below.

Drains are installed in all rooms which could be impacted by a flood. The cable riser galleries have large drainage areas which would hold significant quantities of water. The gaps at the bottom of doors varied between almost nothing to about 3/8". Most of the rooms have multiple doors. While there would be accumulations of water in some areas, there are gaps under the doors leading out to the turbine building area where the water would not cause a problem. Based on the door gap sizes, the safety equipment that could possibly be affected by a flood is limited to the TGIS system (some vulnerable equipment is about 4" off the floor) and the relay cabinets in the relay room (some wiring and relays about 4" off the floor). Loss of the TGIS system would not be a problem during a seismic event since no toxic gases are expected to be released based on the seismic walkdowns and the future strengthening of the ammonia tank (Cycle 9). Piping and nozzles were judged to be able to withstand the seismic event because of adequate supports and inherent ruggedness. The relays which could be affected by a flood could cause a loss of offsite power (which is likely since switchyard capacity is lower than relay chatter capacity), or could cause an inadvertent closure of valves to the standby CCW pump. While the relay chatter could fail the standby pump by blocking suction or discharge, the other two CCW pumps would be unaffected.

Therefore, potential flooding from multiple fire protection system actuations will not significantly impact seismic risk.

NRC Information Notice 94-12 listed six significant insights [3-25, 3-29]. Treatment within the seismic-induced fire and flood evaluation was as follows:

1. Mercury relays in fire suppression (and other) systems - Mercury relays were not identified in the fire suppression or other systems. Chatter of fire suppression actuation relays was evaluated not to be a significant risk contributor.

2. Seismic dust/smoke detectors - Not applicable to SONGS 2/3 as discussed above.

3. Water deluge systems - These potential impacts were identified and evaluated in the section above.

4. Fire suppressant availability during a seismic event - While most plants do not have fire protection systems that are designed for a safe shutdown earthquake, SONGS 2/3 has a specifically designed seismic fire protection water system, with tankers and headers. However, as discussed in the seismic-fire interactions section, there are no fire sources identified at SONGS 2/3 which could fail during a seismic event and impact safety systems.

5. Switchgear fires - Switchgear capacity to withstand seismic events is directly evaluated in the seismic capacity evaluations and includes potential system interactions with nearby equipment.

6. Electro-mechanical components in cable spreading rooms - The seismic capacity walkdown was used to identify and evaluate potential equipment failures in the cable spreading rooms at SONGS 2/3. There were no unanchored electro-mechanical components such as cabinets in these rooms.

Step 3. Evaluation of Core Damage Risk

Based on walkdowns and capacity and impact evaluations, there are no safety equipment that would be impacted by the failure of potential seismic flooding sources. Therefore, there is no core damage risk due to seismic-induced flooding.

3.3.4.4 Conclusions

A systematic evaluation and walkdown of potential seismic-induced fires or floods was performed for the SONGS 2/3 IPEEE. These evaluations included issues such as fires due to potential sources of hydrogen, floods due to multiple actuations of fire suppression systems, and toxic gas release from the ammonia tank. Based on this evaluation, there are no potential seismic-induced fire or flood sources that will affect safety equipment needed for shutdown during or after a seismic event.

TABLE 3.3-1
SEISMIC-INDUCED FIRE SOURCE CHECKLIST

Fire Area	Combustible Type	Combustible Amount	Adequate Capacity (Y/N)	Fire Impact on Equipment (Y/N)	Comment
2-CO-15-1A	Oil	2827 lbs	Y		RCP lube oil and waste lube oil-reviewed during containment walkdown
2-CO-15-1B	Oil	2827 lbs	Y		RCP lube oil and waste lube oil-reviewed during containment walkdown
2-CO-15-167	Oil	7 lbs		N	elevator-no SSC
2-PE-9-2A	Hydraulic Fluid	232 lbs		N	various small sources
2-PE-30-2C	Hydraulic Fluid	232 lbs		N	various small sources
2-PE-63-3B	Alcohol	2 lbs		N	insignificant source
2-SE-(-5)-135B	Oil	15 lbs	Y		CCW pumps
2-SE-(-5)-135C	Oil	15 lbs	Y		CCW pumps
2-SE-(-5)-135D	Oil	15 lbs	Y		CCW pumps
2-SE-(-15)-137C	Oil	44 lbs	Y		SI, LPSI, CS pumps
2-SE-8-140B	Oil	15 lbs		N	valves
2-SE-30-143	Oil	7 lbs		N	elevator-no SSC
2-SE-30-145A	Hydraulic Fluid	1,809 lbs	Y		MSIV, MFIV
2-TB-7-148A	Oil	12,573 lbs		N	no SSC
2-TB-8-148F	Oil	1,110 lbs	Y		in pumps/motors
2-TB-7-150	Oil	7 lbs		N	no SSC
2-DG-30-155	Lube Oil Diesel Fuel	7,200 lbs 4,600 lbs	Y		DG and Fuel Oil
2-DG-30-158	Lube Oil Diesel Fuel	7,200 lbs 4,600 lbs	Y		DG and Fuel Oil
2-TK-30-161A	Oil	984 lbs	Y		drain tanks/lines OK
2-AC-9-8	Oil	12 lbs		N	no SSC -ltg. swgr
2-AC-9-9	Oil	229 lbs	Y		chillers
2-AC-9-10	Oil	229 lbs	Y		chillers
2-AC-9-11	Oil	229 lbs	Y		chillers
2-AC-9-12	Oil	7 lbs		N	no SSC-normal HVAC
2-AC-9-13	Oil	12 lbs		N	no SSC-ltg swgr

TABLE 3.3-1
SEISMIC-INDUCED FIRE SOURCE CHECKLIST

Fire Area	Combustible Type	Combustible Amount	Adequate Capacity (Y/N)	Fire Impact on Equipment (Y/N)	Comment
2-AC-9-16	combustible gases	8 lbs	Y		H2 in cylinder -toxic gas analyzers
2-AC-9-18	Oil	7 lbs		N	elevator-no SSC
2-AC-70-64	combustible gases	4 lbs		N	no SSC-general area
2-AR-9-76	Oil	1 lb		N	no SSC-cabinet xfmr
2-AR-9-84A	Oil	7 lbs		N	BAMU
2-AR-9-84B	Oil	7 lbs		N	BAMU
2-AR-9-87	Oil	75 lbs	Y		Charging pump
2-AR-9-88	Oil	75 lbs	Y		Charging pump
2-AR-9-89	Oil	75 lbs	Y		Charging pump
2-AR-9-90	Oil	7 lbs		N	elevator-no SSC
2-AR-37-102A	Oil	355 lbs		N	no SSC-general area
2-AR-63-116	Oil	104 lbs		N	no SSC-general area
3-AR-9-78A	Oil	7 lbs		N	BAMU
3-AR-9-78B	Oil	7 lbs		N	BAMU
3-AR-9-91	Oil	74 lbs	Y		Charging
3-AR-9-92	Oil	74 lbs			Charging
3-AR-9-93	Oil	74 lbs			Charging
2-YD-30-200A	Diesel Fuel Oil Lube Oil Transformer Oil Snubber Oil Acetylene Paint/Solvent Hydrogen	2,230 lbs 522,870 lbs 863,641 lbs 895 lbs 14 lbs 21,586 lbs 260 lbs	Y	 N N N N N N	 diked transformers-no SSC general-no SSC general bottles general-no SSC H2 tanks and lines included in special walkdown
2-YD-30-200B	Diesel Fuel Oil Lube Oil Transformer Oil Acetone/Solvent Paint EHC Fluid Hydrogen	2,006 lbs 11,477 lbs 566,460 lbs 6,171 lbs 7,624 lbs 5,005 lbs 260 lbs	Y	 N N N N N N	 in equip-no SSC transformers-no SSC general-no SSC general -no SSC general-no SSC H2 tanks and lines included in special walkdown

TABLE 3.3-1
SEISMIC-INDUCED FIRE SOURCE CHECKLIST

Fire Area	Combustible Type	Combustible Amount	Adequate Capacity (Y/N)	Fire Impact on Equipment (Y/N)	Comment
3-CO-15-1A	Oil	2827 lbs	Y		RCP lube oil and waste lube oil-reviewed during containment walkdown
3-CO-15-1B	Oil	2827 lbs	Y		RCP lube oil and waste lube oil-reviewed during containment walkdown
3-CO-15-167	Oil	7 lbs		N	elevator-no SSC
3-PE-9-2A	Hydraulic Fluid	232 lbs		N	various small sources
3-PE-30-2C	Hydraulic Fluid	232 lbs		N	various small sources
3-PE-63-3B	Alcohol	2 lbs		N	insignificant source
3-SE-(-5)-135B	Oil	15 lbs	Y		CCW pumps
3-SE-(-5)-135C	Oil	15 lbs	Y		CCW pumps
3-SE-(-5)-135D	Oil	15 lbs	Y		CCW pumps
3-SE-(-15)-137C	Oil	44 lbs	Y		SI, LPSI, CS pumps
3-SE-8-140B	Oil	15 lbs		N	valves
3-SE-30-142A	Oil	28 lbs		N	elevator-no SSC
3-SE-30-145A	Hydraulic Fluid	1,809 lbs	Y		MSIV, MFIV
3-TB-7-148A	Oil	13,134 lbs		N	no SSC
3-TB-8-148F	Oil	1,110 lbs	Y		in pumps/motors
3-TB-7-150	Oil	7 lbs		N	no SSC
3-DG-30-155	Lube Oil Diesel Fuel	867 gal 550 gal	Y		DG and Fuel Oil
3-DG-30-158	Lube Oil Diesel Fuel	867 gal 550 gal	Y		DG and Fuel Oil
3-TK-30-161A	Oil	984 lbs	Y		drain tanks/lines OK

TABLE 3.3-2
SEISMIC-INDUCED HYDROGEN FAILURE CHECKLIST

Fire Area or General Area	Room	Pipe	Adequate Capacity (Y/N)	Impact on Equipment (Y/N)	Comment
2-PE-9-2A	109	110-1"-D-LLO		N	In Rad Waste Pipe Chase, 2-AR-24-94, piping but no SSC
2-PE-30-2D	209	098-3/4"-J-KEO	Y		
	209	066-3'-D-LLO	Y		
3-PE-9-2A	109	110-1"-D-LLO		N	In Rad Waste Pipe Chase, 2-AR-24-94, piping but no SSC
3-PE-30-2D	209	098-3/4"-J-KEO	Y		
	209	066-3"-D-LLO	Y		
Outside		piping	Y	N	piping supported OK and no SSC impacts
Yard		storage tanks		N	no SSC
General		H2 cylinders	Y		anchored or chained adequately
Turbine Building		H2 storage		N	no SSC
Rad Waste Bldg		piping		N	no SSC in areas with H2 piping

Note: If a fire or flood source has adequate capacity, fire or flood impact was not assessed. If a fire or flood source has no fire or flood impact, seismic ruggedness was not evaluated.

TABLE 3.3-3
SEISMIC-INDUCED FLOOD SOURCE CHECKLIST

Flood Area	Flood Source or Potential Impact Area	Adequate Capacity (Y/N)	Flood Impact on Equipment (Y/N)	Comment
Plant Grade	Fire Water Tanks T102, T103		N	
	Demin Water Storage Tanks		N	graded away from SSC
	Turbine Plant CW HXs/Pumps		N	
	Holdup Tank T-258		N	
	Turbine Plant CW Surge Tank T-50		N	
	Sulfuric Acid Tank T-194		N	diked
	Ammonia Tank T-106	Y		contingent on upgrade
Plant Intake Structure	Circ Water Pumps	Y		
	SWC Pumps	Y		
	Screen Wash Pumps	Y	N	drains down
Turbine Bldg	Condenser Expansion Joints		Y	
Control Bldg-9'	Chillers	Y		
	Relay Room		Y	deluge systems could flood area -relays about 4" above floor
Control Bldg-30'	Control Room		N	no significant flood source
	Deluge Valves		N	flood in cable riser gallery would drain to 9' level
Control Bldg-50'	Switchgear, Inverter, Battery Rooms		N	no significant flood source
	Deluge Valves		N	flood in cable riser gallery would drain to 9' level
Safety Equipment Bldg-(-15', -5', 9')	Piping	Y		
	CCW Surge Tanks	Y		
AFW Pump Room	Piping	Y		
	Deluge Valve		N	TDAFWP and Tr B equipment is qualified for spray effects
	Nuclear SW Storage Tank T-104		N	bermed
Doghouse	Piping	Y		
RadWaste Bldg	Primary Plant Makeup Tanks		N	walled

TABLE 3.3-3
SEISMIC-INDUCED FLOOD SOURCE CHECKLIST

Flood Area	Flood Source or Potential Impact Area	Adequate Capacity (Y/N)	Flood Impact on Equipment (Y/N)	Comment
DG Bldg	Piping	Y		
General	Hill Tanks		N	Diked. Slopes & grade such that H20 would drain to ocean. Bldgs doors prevent significant flooding.

Note: If a fire or flood source has adequate capacity, fire or flood impact was not assessed. If a fire or flood source has no fire or flood impact, seismic ruggedness was not evaluated.

3.3.5 UNIT 2 - UNIT 3 COMPARISON

The walkdowns of both units confirmed that there are only minor differences between Unit 2 and Unit 3. The differences do not affect the seismic fragility of the structures, equipment or components that were designated as essential in the IPEEE project.

3.3.6 RECOMMENDED CORRECTIVE ACTIONS

As a result of the walkdowns, several seismic interaction concerns were identified. The seismic interactions pertain to closely-spaced electrical panels with essential relays, proper restraints for overhead light fixtures, and small gaps between valve operators and floor gratings. These items will be modified to eliminate the potential seismic interaction conditions. The modifications are summarized in Section 3.7.

3.4 ANALYSIS OF PLANT SYSTEMS AND STRUCTURAL RESPONSE

A probabilistic response analysis of structures for SONGS was performed in support of the seismic IPE program. In general, floor response spectra and structure member forces developed for the plant's design basis are considered to be conservatively biased. Hence, it was decided to generate new structural responses using the current state-of-the-art methods and avoid any intentional bias in the analysis with respect to soil-structure modeling. In order to cast the results in a form convenient for the development of structure and equipment fragilities, a probabilistic approach was adopted.

The objectives were twofold:

- To estimate median structure forces and the variability about the median for all major structures of interest for input to structure fragility analysis.

- To develop probabilistic floor response spectra in all major structures of interest for use in equipment fragility analysis

The following Seismic Category I structures were analyzed and evaluated:

Auxiliary Building
Containment/Internal Structure
Safety Equipment Building
Diesel Generator Building
Condensate and Refueling Tank Enclosure Building
Intake Structure

3.4.1 METHODOLOGY

Seismic IPEEE methods require that the best estimate or median-centered seismic response of buildings and equipment be evaluated for earthquake well beyond the design basis earthquake (DBE). The term "median-centered response" denotes best estimate or 50% non-exceedance probability (NEP) response conditional on the occurrence of the ground motion.

Probabilistic seismic response analysis is generally used to calculate seismic responses for use in a seismic PRA. Further, the location of the SONGS buildings on a deep soil site requires the consideration of soil-structure interaction (SSI) effects in the response analyses. The basic elements of SSI analysis and the probabilistic response analysis are described in Appendices A and B of Reference [3-33], respectively. The overall approach is described in the context of the substructure method. The elements of the substructure approach as applied to structures subjected to earthquake excitations are:

- Specifying the free-field ground motion.

- Development of the soil models, i.e., defining the soil profile and performing the site response analysis.

- Calculating the foundation impedance functions and wave scattering effects.

- Determining the fixed-base dynamic characteristics of the structure.
- Performing the SSI analysis, i.e., combining the previous steps to calculate the response of the coupled soil-structure system.

The approach to probabilistic response analysis is to perform multiple SSI analyses using the above methodology. For each earthquake simulation, variations in input motions and SSI parameters (structural frequency and damping, and soil shear modulus and damping) are introduced. The end products of a probabilistic response analysis are distributions on structure response -- i.e., loads in structural elements for structure fragilities, and in-structure response spectra which define the seismic demand on equipment housed in the buildings. The distributions are described by the 50th percentile values and the logarithmic standard deviations (β's).

In past applications, probabilistic response analyses were performed for a number of different free-field ground motion levels. These levels were conveniently selected at multiples of the Safe Shutdown Earthquake (SSE) level, and the acceleration levels at which failures were estimated to occur were obtained by interpolation, or more commonly by extrapolation, of the analytical data. Note that the term "SSE" is analogous to "DBE". In this report, "DBE" is used to denote the design basis ground input. "SSE" is used to denote earthquake excitation levels in the probabilistic response analysis. Direct scaling of results from one earthquake level to another is not strictly correct due to nonlinearity in soil behavior. For SONGS, the level at which structures and equipment are expected to be challenged is anticipated to be twice the SSE (2xSSE) level. The SSE level is also analyzed to provide a data point for interpolation/extrapolation as well as a reference point for comparison against the original design basis results.

3.4.1.1 Free-field Ground Motion

In probabilistic response analysis, the characteristics of the free-field ground motion is defined by the shape of the mean uniform hazard spectra (UHS) corresponding to a return period of interest. The UHS shape is a product of a probabilistic seismic hazard analysis. In general, UHS shapes are defined for both the horizontal and vertical directions. However, due to time constraints, it was necessary to start the probabilistic response analysis prior to completion of the probabilistic seismic hazard analysis task. To permit this parallel effort, a preliminary estimate for the shape of the UHS in the horizontal and vertical directions was made by Woodward-Clyde Consultants (WCC).

To complement the UHS, an ensemble of 26 time histories consistent with the preliminary UHS was provided by WCC [3-34]. Table 3.4-1 lists the source of this strong motion ensemble. For each time history in the ensemble, the horizontal components were normalized such that the median spectral acceleration of the average of the two horizontal components over the frequency band of 1 to 10 Hz was unity, using even sampling along the frequency axis. The vertical component was normalized by the median vertical spectral acceleration over the frequency band of 1 to 10 Hz, and then scaled by the average ratio of the vertical to horizontal uniform hazard normalized spectral shape over 1 to 10 Hz. In the probabilistic response analyses, these normalized time histories were further scaled to the SSE and 2xSSE level in the following manner. First, the average spectral acceleration from 1 to 10 Hz was computed for the SSE spectral shape. This involved digitizing the SSE at the frequency set {f=1, 2.5, 5, and 10 Hz} and applying the following equation:

$$S_n = (0.5^*S_{a,1Hz} + S_{a,2.5Hz} + S_{a,5Hz} + 0.5^*S_{a,10Hz}) / 3$$

Using the above, the average spectral acceleration computed for the SSE was 1.43g. Thus, the normalized time histories were scaled by a factor of 1.43 to yield the SSE level because the time histories were developed for an average spectral acceleration of 1.0g. Similarly, the 2xSSE level was obtained by applying a scale factor of 2.86. These scale factors were applied to each time history in the normalized ensemble discussed in the preceding paragraphs. Reference [3-35] describes the time history ensemble and the scale factors in detail.

Figures 3.4-1 through 3.4-3 compare the recommended UHS scaled to the SSE level against the DBE. It may be seen that in the horizontal direction, the UHS shape is below the DBE at frequencies below 2 Hz. Hence soil-structure modes below 2 Hz will experience lower seismic input using the UHS. The vertical UHS exhibits a significant frequency shift to peak at around 10 Hz (See Figure 3.4-3). This frequency shift is expected to propagate into the vertical floor response spectra. Also plotted in Figures 3.4-1 through 3.4-3 are the 50% and 84% non-exceedance probability (NEP) response spectra computed from the time history ensemble. The 84% NEP response spectra provide an idea of the variability of the time history ensemble about the median. The 50% and 84% NEP response spectra of the ensemble are computed with no assumption about the distribution of the data.

In order to validate the use of the preliminary UHS and the associated time history ensemble, the preliminary UHS recommended by WCC were compared against the final UHS from the seismic hazard analysis conducted by Risk Engineering, Inc (REI). Figure 3.4-4 compares the preliminary horizontal UHS recommended by WCC against REI's UHS at 1.386×10^{-4} exceedance probability (SSE level). Both spectral shapes have been normalized over the 1 to 10 Hz range. It may be seen that the preliminary estimate matches REI's final UHS shape extremely well. Considering the overriding importance of the horizontal direction with respect to structure and equipment fragility, a further check was performed at the 2xSSE level on the horizontal spectra. It may be seen in Figure 3.4-4 that the 2xSSE and SSE shapes from the seismic hazard analysis are very similar, thereby validating the use of preliminary UHS shapes recommended by WCC. In the vertical direction, the vertical UHS recommended by WCC shows some conservatism over REI's UHS at the SSE level as shown in Figure 3.4-5. The general shape, particularly the higher frequency content as compared to the horizontal UHS shape, are in good agreement.

TABLE 3.4-1
EMPIRICAL TIME HISTORIES SELECTED FOR PROBABILISTIC RESPONSE ANALYSES
(Extracted from Reference [3-1])

Simulation Number	Earthquake	Station	Magnitude	Distance	Mechanism	Site Condition
1	1940 Imperial Valley	ICSB	7.1	8.3	Strike-Slip	Soil
2	1978 Tabas	Tabas	7.4	3.0	Reverse	Soil
3	1979 Imperial Valley	Aeropuerto	6.5	5.2	Strike-Slip	Soil
4	1979 Imperial Valley	Agrarias	6.5	5.5	Strike-Slip	Soil
5	1979 Imperial Valley	Brawley	6.5	8.5	Strike-Slip	Soil
6	1979 Imperial Valley	El Centro #10	6.5	8.6	Strike-Slip	Soil
7	1979 Imperial Valley	Holtville PO	6.5	7.5	Strike-Slip	Soil
8	1983 Coalinga	PVPP	6.5	8.5	Reverse	Soil
9	1987 Superstition Hill (B)	Westmoreland	6.7	13.4	Strike-Slip	Soil
10	1989 Loma Prieta	Capitola	7.0	14.5	Oblique	Soil
11	1989 Loma Prieta	Gilroy #2	7.0	12.7	Oblique	Soil
12	1989 Loma Prieta	Gavilian College	7.0	11.6	Oblique	Soil
13	1992 Landers	Joshua Tree	7.3	12.0	Strike-Slip	Soil
14	1992 Petrolia	Petrolia FS	7.1	10.0	Reverse	Soil
15	1992 Petrolia	Rio Dell	7.1	14.7	Reverse	Soil
16	1971 San Fernando	Pacoima Dam	6.6	2.8	Reverse	Rock
17	1976 Gazli	Gazli	6.8	3.0	Reverse	Rock
18	1989 Loma Prieta	Corralitos	7.0	5.1	Oblique	Rock
19	1966 Parkfield	Chalome #5	6.1	5.3	Strike-Slip	Soil
20	1966 Parkfield	Chalome #8	6.1	9.2	Strike-Slip	Soil
21	1972 Managua	Esso	6.2	5.0	Strike-Slip	Soil
22	1980 Mexicali	Chihuahua	6.4	14.6	Strike-Slip	Soil
23	1980 Mammoth Lakes (A)	Convict Creek	6.2	15.0	Strike-Slip	Soil
24	1980 Mammoth Lakes (A)	Mammoth H.S.	6.2	14.0	Strike-Slip	Soil
25	1984 Morgan Hill	Gilroy #4	6.2	12.8	Strike-Slip	Soil
26	1984 Morgan Hill	Halls Valley	6.2	3.4	Strike-Slip	Soil

3.4.1.2 Soil Modeling

The native soil at the site consists of about 70 feet of terrace deposit (from Elevation 120' to Elevation 50'), underlain by approximately 900 feet of San Mateo sand. The San Mateo Formation is a very dense well graded sand which exhibits some apparent cohesion and high shear strength due to efficient grain packing. The San Mateo formation at the site is quite uniform, with no significant continuous layering. In the plant area, the top 70 feet of terrace deposit was completely removed, and plant grade is established at about Elevation 30'. The maximum groundwater table at the site is at Elevation 5', i.e, about 25 feet below plant grade. All Seismic Category I structures are founded on the San Mateo sand.

The shear wave velocity increases with depth as a result of increased confinement. In developing the low strain soil profile, the effect of the structural weight on the effective confinement pressure was considered. For near surface soil (i.e., upper 15 feet), the average shear wave velocity measured was 930 fps. For the San Mateo Sand, the shear modulus is related to the effective confinement by the following relation:

$$G = 100\, K_m \sigma_m^{2/3} \qquad \text{(Eqn 3-2)}$$

where σ_m = $2/3\ \sigma_o$

σ_o = Overburden pressure in units of lbs/ft^2

K_m = Dimensionless soil modulus coefficient which is a function of void ratio and shear strain amplitude

Using equation 3-2, the low strain shear wave velocity profiles in the free-field and under the various structures were computed and plotted in Figure 3.4-6. Included in the calculation of effective confinement was the buoyancy of material below the water table. The buoyancy effect tended to reduce the effective confinement and led to decreased shear moduli. The major structures are sited close together, resulting in overlapping of stress fields generated by the foundation pressures. Therefore at some depth away from the foundations, the distinction between the various soil profiles shown in Figure 3.4-6 diminishes. Note that a shear wave velocity of 930 fps is used for the near surface soil, based on Rayleigh wave measurements.

For the purpose of the probabilistic response analysis, it was deemed sufficient to use a single best estimate soil profile for the various structures, and to treat the effect of the different structural bearing pressures as part of the uncertainty in modeling the soil stiffness.

3.4.1.3 Dynamic Soil Properties

Dynamic material properties of the foundation soils were determined from existing field seismic surveys and laboratory cyclic triaxial tests. The effect of soil strain level and degree of confinement on its shear modulus and hysteretic damping are summarized in Figure 3.4-7.

3.4.1.4 Strain Compatible Soil Properties

Site response analyses were performed for the SSE and 2xSSE earthquake levels to establish median high strain soil properties. The computer program SHAKE [3-36] was used for all analyses. SHAKE is typically used to perform one-dimensional wave propagation analyses using soil columns to represent the site profile. The site response analyses for SONGS used the following input:

- *Soil column with low strain soil properties* - The best estimate low strain soil profile is based on the free-field profile shown in Figure 3.4-6
- *Dynamic soil properties* - Dynamic soil property curves for the San Mateo sand are shown in Figures 3.4-7 and 3.4-8.
- *Horizontal component of ground motion* - The ground motion in this case is an artificial time history that matches the SONGS 2&3 mean horizontal UHS.
- *Control point location* - The mean UHS is specified in the free-field at plant grade (Elevation 30'), hence the control point is Elevation 30' of the free-field soil column.

The input motion was scaled to the SSE level and iterations were performed by SHAKE to obtain strain-compatible properties. The process was repeated for the 2xSSE level with the input motion scaled to a higher level. Reference [3-37] documents the SHAKE analyses.

3.4.1.5 Foundation Impedances and Wave Scattering Functions

The substructure approach to SSI analysis was used in this study and one element of this approach is the development of foundation impedances and wave scattering functions.

The foundation impedances describe the harmonic force-displacement characteristics of the soil and are dependent on soil configuration and material behavior, the frequency of excitation, and the geometry of the foundation. For the assumption of a rigid foundation, the force-displacement characteristics are uniquely defined by a 6 x 6 matrix relating a resultant set of forces and moments to the six rigid-body degrees-of-freedom. In general, for linear elastic or viscoelastic material and a uniform or horizontally stratified soil deposit, each element of the matrix, $[K_s(\bar{\omega})]$, is complex-valued and frequency-dependent. Each complex element of the matrix can be thought of as a pair of functions: the real part approximating the stiffness of the soil, and the imaginary part representing the damping. For surface foundations at SONGS, the computer program CLASSI [3-38] was used for calculating foundation impedances. For embedded foundations, the approach was to first compute surface foundation impedances using CLASSI, followed by applying a correction for embedment. The computer program CYLREC [3-39] was used to compute the correction terms. The soil properties used as input to impedance calculation were based on the best estimate soil profile compatible with the SSE-induced strain level.

The scattering matrix $[S(\bar{\omega})]$, relates the foundation input motion $\{U^*(\bar{\omega})\}$, to the surface free-field ground motion according to the following transformation:

$$\{U^*(\bar{\omega})\} = [S(\bar{\omega})]\,\{f(\bar{\omega})\}$$

The vector $\{f(\bar{\omega})\}$ is the complex Fourier transform of the free-field surface ground motion. Thus, applying the scattering matrix to the free-field surface ground motion yields the foundation input motion. The foundation input motion differs from the free-field ground motion in all cases, except for surface foundations subjected to vertically incident seismic waves. First, the free-field motion varies with soil depth. Second, the soil-foundation interface scatters waves because points on the foundation are constrained to move according to its geometry and stiffness. For vertically propagating seismic waves impinging on surface foundations, the foundation input motion is the same as the free-field motion. Wave scattering functions for the embedded foundations at SONGS were computed using computer program SASSI developed at the University of California, Berkeley [3-40, 3-41]. The wave scattering calculations used 2-D plane strain models subjected to vertically incident shear and compressional waves. In general, two plane strain models per structure are required to represent vertical slices in the N-S and E-W directions. If the foundation dimensions and embedment conditions are similar for the N-S and E-W slices, then only one plane strain model is sufficient.

Details of the foundation models used for computing impedances and wave scattering functions are described in Reference [3-33]. Foundation impedances and scattering functions were explicitly computed only for the SSE soil profile. The 2xSSE case was obtained by scaling the SSE soil shear modulus and damping by average factors to account for soil degradation at higher strain levels.

3.4.1.6 Structure Models

The SSI analyses for SONGS 2&3 utilized the substructure approach as described in Appendices A and B of Reference 3-33. The structure models required for this approach are fixed-base and SSI effects are incorporated using foundation impedances and wave scattering functions. Structure models developed for the original design analyses and reported in the FSAR are representative of current procedures, and are considered as best-estimate models for the purpose of this study. These structure models typically consist of lumped masses interconnected by beams or stiffness matrices, and are three dimensional to capture any torsional effects. The original analyses treated SSI by the "lumped parameter" approach. Therefore, the FSAR models invariably include soil springs to represent the flexibility of the supporting media. These original models are in the format of either one of two structural analysis computer programs used by Bechtel Corporation, called BSAP (a derivative of SAPIV) and SUPERSMIS. SUPERSMIS uses 12x12 element stiffness matrices to represent the interconnecting spring elements.

The fixed-base structure models were first reconstructed in either MODSAP [3-42,3-43] or SUPERSAP [3-44] format, based on the original FSAR models. Wherever possible, these reconstructed models initially retained the original soil springs for benchmarking against the FSAR model. Benchmarking was accomplished by comparing the computed natural frequencies. Tables 3.4-2, 3.4-3, and 3.4-4 illustrate the natural frequency comparisons for the auxiliary building, containment structure, and diesel generator building models. The fixed-base models were then readily obtained by removing the soil springs. The FSAR models were reconstructed with essentially no changes to their dynamic characteristics. In order to capture the effects of foundation rocking on vertical responses, and foundation torsion on translational responses, additional massless nodes were added at the corners of floor slabs. At any given floor, these nodes were rigidly linked to the center of mass of the floor slab.

In the case of the safety equipment building, the FSAR analyses used a large plates/shell finite element model. This model was considered to be too elaborate for the purpose of floor response spectra generation. Hence, a simpler stick model was developed based on structural drawings.

TABLE 3.4-2
DYNAMIC CHARACTERISTICS OF FIXED-BASE AUXILIARY BUILDING MODEL
Percent Mass Participation

Mode #	Freq of Current Model (Hz)	X	Y	Z	XX	YY	ZZ	Freq of Orig Fixed Base Model
1	10.14	65.75	1.38	0.00	0.82	39.48	11.98	10.14
2	10.36	1.30	83.34	0.10	51.35	0.80	0.00	10.36
3	12.10	15.60	0.06	0.00	0.05	9.61	72.79	12.10
4	20.76	0.03	8.07	8.11	2.70	0.03	0.08	20.76
5	22.24	7.99	0.03	2.64	0.10	6.85	0.78	22.24
6	22.73	0.46	0.03	48.49	3.12	0.00	0.56	22.73
7	23.33	0.00	0.00	0.03	0.04	0.01	0.07	23.32
8	24.02	0.79	0.01	0.57	0.07	9.89	1.69	24.01
9	24.80	0.06	0.73	2.31	0.07	0.09	5.15	24.80
10	26.94	0.64	0.00	0.06	0.07	23.03	0.20	26.93
11	29.33	0.00	2.18	15.71	18.03	0.03	0.30	29.33
12	31.89	0.01	2.03	5.90	15.17	0.00	0.08	31.89
13	32.74	3.10	0.02	0.01	0.00	0.07	0.42	32.74
14	36.11	1.08	0.00	0.02	0.03	0.06	3.39	36.11
15	43.13	0.00	0.70	0.00	0.03	0.00	0.04	43.12
Total % Mass		96.81	98.57	83.94	91.66	89.94	97.54	Ref. [3-45]

TABLE 3.4-3
BENCHMARKING OF CURRENT MODEL AGAINST FSAR MODEL FOR CONTAINMENT/INTERNAL STRUCTURE (WITH SOIL SPRINGS)

Mode #	Frequencies (Hz)		Mode #	Frequencies (Hz)	
	Current Model	FSAR Model		Current Model	FSAR Model
1	1.70	1.70	26	21.21	21.21
2	1.71	1.71	27	21.52	21.60
3	2.56	2.56	28	21.87	21.92
4	2.63	2.63	29	21.99	22.01
5	2.63	2.63	30	22.05	22.13
6	3.35	3.35	31	22.19	22.28
7	3.37	3.37	32	22.36	22.40
8	8.51	8.52	33	22.38	22.42
9	8.60	8.60	34	22.44	22.60
10	9.70	9.74	35	22.73	22.88
11	11.24	11.24	36	22.98	23.28
12	11.78	11.85	37	23.51	23.78
13	12.95	13.00	38	24.50	24.56
14	13.96	14.00	39	29.54	30.36
15	15.92	15.97	40	30.03	31.47
16	16.78	16.79	41	31.39	31.75
17	17.22	17.23	42	31.57	31.96
18	17.23	17.56	43	31.80	32.01
19	18.19	18.29	44	31.92	32.41
20	18.56	18.69	45	32.13	32.78
21	18.57	18.72	46	32.72	33.48
22	18.66	18.85	47	33.77	34.24
23	19.17	19.56	48	35.34	35.94
24	20.28	20.29	49	35.96	35.98
25	21.07	21.08	50	36.01	36.15

TABLE 3.4-4
BENCHMARKING OF CURRENT MODEL AGAINST FSAR MODEL FOR DIESEL GENERATOR BUILDING (WITH SOIL SPRINGS)

Mode	Frequencies (Hz)	
	Current Model	Original Model
1	3.37	3.38
2	3.38	3.39
3	4.14	4.14
4	4.25	4.26
5	5.78	5.78
6	6.37	6.38
7	28.67	28.73
8	33.57	33.59
9	34.49	34.63
10	42.22	42.23

3.4.1.7 Probabilistic Response Analyses

In the final step of the probabilistic response analyses procedure, the various elements of the substructure approach previously discussed were combined to calculate probability distribution on structure loads and in-structure response spectra. The probabilistic approach is based on work performed under the Seismic Safety Margins Research Program (SSMRP [3-45]) and it involves multiple earthquake simulations. For SONGS, 26 deterministic SSI analyses were performed using the free-field ground motions described in Section 3.4.1.1. For each analysis or earthquake simulation, key structure and soil parameters were randomly sampled from assumed lognormal distributions. The sampling procedure was based on the Latin Hypercube experimental design described in Reference [3-46]. The parameters that were varied during the SSI simulations, and the assumed coefficient of variation (COV) of the lognormal distributions for each parameter are:

SSI PARAMETER	COV
Soil Shear Modulus	0.50
Soil Damping	0.60
Structural Frequency	0.25
Structural Damping	0.35

These assumed COV's are based on previous work and expert judgement. They include all modeling and random uncertainty in the estimation of the best estimate values. The most important parameter for the SONGS analyses is the soil shear modulus variability. Soil test data was reviewed and expert opinion applied to estimate the variability for this parameter. Table 3.4-5 lists the experimental design generated using the above COV's for each parameter.

TABLE 3.4-5:
LATIN HYPERCUBE EXPERIMENTAL DESIGN

Simulation #	Soil Shear Modulus	Soil Damping	Structural Frequency	Structural Damping
1	0.41531	0.24387	0.55858	0.53840
2	0.44748	0.37700	0.65849	0.59593
3	0.52998	0.48711	0.73401	0.64530
4	0.61121	0.56221	0.77319	0.68324
5	0.65568	0.60170	0.78828	0.71011
6	0.68080	0.64100	0.8152	0.76622
7	0.72397	0.69238	0.85286	0.80059
8	0.77804	0.72565	0.88301	0.83650
9	0.79807	0.77455	0.89955	0.86135
10	0.83990	0.82789	0.91191	0.88341
11	0.89453	0.85040	0.94433	0.92129
12	0.92744	0.93574	0.97451	0.93873
13	0.99229	0.96678	0.98536	0.99246
14	1.00070	1.04790	1.02154	1.00973
15	1.06045	1.05574	1.04624	1.06020
16	1.10800	1.14213	1.05048	1.08990
17	1.17232	1.24214	1.08028	1.11029
18	1.22003	1.24881	1.12966	1.18234
19	1.30174	1.39971	1.14352	1.21636
20	1.41034	1.42538	1.17069	1.26241
21	1.43055	1.59621	1.19962	1.32435
22	1.58448	1.68884	1.25786	1.37524
23	1.75420	1.79050	1.29714	1.49184
24	1.95477	2.20479	1.3495	1.61658
25	2.19660	2.27749	1.49709	1.77849
26	2.76170	3.36624	1.67739	2.02865

3.4.2 RESULTS FOR STRUCTURES

As discussed in Section 3.4.1.1, the structures were analyzed for two earthquake levels: SSE and 2xSSE. The response quantities of interest recovered from the multiple earthquake simulations included peak accelerations, maximum member forces, and floor acceleration time histories. These quantities were needed for fragility development. In addition, maximum foundation loads were also computed. Floor acceleration time histories computed for each of the 26 simulations were post-processed into 5% damped floor response spectra. For each location, the spectral accelerations were arranged in ascending order and the median and 84th percentile values extracted. In a similar manner, maximum member forces were recovered and post-processed into median and 84th percentile values. The results were documented in Reference [3-33]. Tables of maximum response quantities, and plots of 5% damped floor response spectra are contained therein. The plots display smoothed median spectral values as well as smoothed logarithmic standard deviations.

The comparison of the current median-centered floor response spectra against the original FSAR results indicates the level of conservatism in the FSAR analysis. The comparison was performed for the auxiliary building and the diesel generator building. The auxiliary building is large and houses many critical equipment items, and is embedded to some extent. The diesel generator building (DGB) represents the other extreme; it is light and founded on the surface. Hence, SSI effects on these two structures are indicative of trends for the other SONGS structures.

Figures 3.4-9, 3.4-10, and 3.4-11 compare responses at the basemat of the auxiliary building. The lumped parameter approach of the FSAR analysis restricted soil material and radiation damping to 10% of critical even though the theoretical damping of 46% for the translational mode was computed. In the current analysis using impedance functions, no restriction was placed on the soil damping, which is calculated to be about 52% of critical for the fundamental coupled translation/rocking mode. It is observed in Figures 3.4-9 and 3.4-11 that the soil-structure system is very effective in filtering out high frequency contents in the free-field ground motion. This is to be expected for a highly damped soil-structure system with fundamental system frequency less than 2 Hz. Using the median-centered approach, a ground excitation level of twice the SSE resulted in structural responses that are lower than the original DBE floor spectra for frequencies below 2.5 Hz. At higher frequencies, the 2xSSE median response spectra are only slightly higher than the original design basis. In an approximate sense, the median-centered approach almost reduces the FSAR results by 50% in the horizontal direction. However, in the vertical direction, a different scenario is observed. Due to the shift in frequency content of the vertical UHS ground spectrum, a corresponding shift in the basemat response spectrum is seen (see Figure 3.4-11). As a result, the vertical

2xSSE median spectrum is well above the original design basis at higher frequencies. The above trends hold true for higher elevations within the auxiliary building (see Figures 3.4-12, 3.4-13, and 3.4-14 for El. 63.5').

For the surface founded DGB, the foundation did not filter the free field input motion as effectively as the auxiliary building. It may be seen in Figures 3.4-15 and 3.4-16 that the horizontal free-field input motion is transmitted to the basemat of the DGB almost unfiltered. For this building, the 2xSSE median response spectra in the horizontal directions are generally somewhat higher than the original design basis. In the vertical direction (see Figure 3.4-17), the shift in floor response spectra to higher frequency is once again observed. The DGB basemat is more effective in attenuating the high frequency content in the vertical ground input, again due to high soil damping for the vertical mode.

For the other embedded structures (containment, safety equipment building, intake structure), the SSI effects are more pronounced than for the surface founded structures.

3.5 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

3.5.1 METHODOLOGY

Seismic fragilities of structures and equipment were estimated using the methods described in EPRI "Methodology for Developing Seismic Fragilities," [3-47]. The seismic fragilities were developed in terms of the average spectral accelerations in the 1 to 10 Hz frequency range. The three fragility parameters $\hat{A}$, β_R, and β_U have been calculated for various components deemed to be critical and most vulnerable to failure during a seismic event.

3.5.1.1 Structure Fragilities

Seismic fragilities of important structures were estimated using the results from the probabilistic response analysis described in Section 3.4. Structures are considered to fail functionally when inelastic deformations of the structure under seismic loads are sufficient to interfere with the operability of safety-related equipment attached to the structure. These limits on inelastic energy absorption capability (ductility limits) chosen for structures are estimated to correspond to the onset of significant structural damage. For each structure, the seismic fragilities are described in terms of the median 5 percent damped free field spectral acceleration $\hat{A}$, and random and uncertainty logarithmic standard deviations β_R, and β_U. The factor of safety is defined as the ratio of the free field spectral acceleration capacity, to the 2XSSE reference earthquake acceleration used in the structure response analyses [3-33].

The factor of safety for the structure capacity F_{cap} consists of the following parts:

1. The strength factor, F_S, based on the ratio of actual member strength to the reference earthquake forces.

2. The inelastic energy absorption factor, F_μ, is related to the ductility of the structure and to the earthquake magnitude range that is believed to contribute most to the seismic risk.

The factor of safety, F_R, related to building response was determined from a number of variables which include:

1. The response spectra used as the basis for analysis.

2. Damping used in the analysis compared with damping expected at failure.

3. Modal combination methods.

4. Combination of earthquake components.

5. Modeling accuracy.

6. Soil-structure interaction effects.

Median factor of safety, F, and variability, β_R and β_U, estimates were made for each of the parameters affecting capacity and response. These median and variability estimates were then combined using the properties of the lognormal distribution to obtain the overall median factor of safety and variability estimates required to define the fragility curve for the structure.

3.5.1.2 Equipment Fragilities

Equipment seismic fragilities were estimated using the methods described in EPRI's report on the "Methodology for Developing Seismic Fragilities" [3-47]. Fragility analyses were performed on equipment that were qualified either by testing or analysis.

3.5.1.3 Relay Fragilities

Relay fragilities were calculated by using relay test results and Generic Equipment Ruggedness Spectra (GERS) from EPRI. Appropriate in-cabinet amplifications were generally determined from specific equipment tests. When test data was not available, generic amplifications were used or specific factors were calculated.

3.5.2 SCREENING RESULTS

A large number of components has been determined by earthquake experience to be seismically rugged when properly anchored and when other criteria are met. For example, valves such as motor-operated valves have been found to have high seismic capacity if the associated piping and valve are properly supported and when the weight of the operator, operator length, and pipe diameter relations are kept within certain experience parameters. These caveats and anchorage criteria are specified in EPRI's Seismic Margin Methodology [3-24] and the seismic capacity walkdown used approved worksheets to verify these criteria are met. Based on walkdown findings and seismic capability engineering evaluations, many of the components were screened out from further fragility analysis because they have high seismic capacity. These screened out components are denoted by an "S" in the Spectral Acceleration (SA) column in Table 3.11-1.

3.5.3 DETAILED FRAGILITY RESULTS

The seismic fragilities are summarized in the Seismic Equipment List in Section 3.11.

3.5.4 SOIL LIQUEFACTION

The potential for ground (soil) failure was evaluated through the collection and review of pertinent documents and re-analysis for high ground acceleration shaking levels. The analysis focused on four key soil-related issues:

1. Ground failure of San Mateo sand in plant area;

2. Liquefaction of filled cavities adjacent to or beneath important structures that formed in the San Mateo sand during dewatering;

3. Offshore conduit blockage due to conduit separation and backfill inflow caused by buoyancy during liquefaction of dumped fill; and

4. Cut slopes in native soil deposits adjacent to critical plant facilities.

The conclusions are summarized as follow:

1. Ground Failure. Ground failure, including liquefaction, bearing failure, lateral spread, or earthquake-induced settlement, is not expected to occur in the San Mateo sand at the 5.4g average spectral acceleration ground shaking level and would not likely occur at greater shaking levels. The San Mateo sand is considered to be a dilative material within the range of depths considered to be important in the plant area. The potential for ground failure was evaluated by considering the material behavior of the sand based on analysis of measured in situ densities, measured blowcounts, cyclic laboratory strength tests, and the results of other studies. These data were also compared with findings from the response of sands during earthquakes and the results of other laboratory studies were used as supporting evidence. Specifically:

   - The range of strain-compatible shear moduli at the 5.4g average spectral acceleration shaking level were found to fall within the range used in the SSI as shown on Figure 3.5-1.

   - The distribution of dry densities indicates that 95 percent of the in situ densities have a relative density of 100 percent (or greater) and are over 100 percent modified relative compaction as shown on Figure 3.5-2. At this density, the limiting (maximum) shear strains are expected to be negligible as shown on Figure 3.5-3.

   - The minimum $(N_1)_{60}$ is 49 and 50 percent of blowcounts exceed 100 blows per foot as shown on Figure 3.5-4. The full range of $(N_1)_{60}$ are well beyond the threshold for liquefaction as shown on Figures 3.5-5 and 3.5-6. The volumetric strain is also negligible for this range of $(N_1)_{60}$ as shown on Figure 3.5-7.

   - The results of stress- and strain-controlled cyclic triaxial tests sampled San Mateo sands are consistent with the results of other studies and show that the native San Mateo sand under in situ conditions will have a negligible limiting strain as inferable from Figures 3.5-8 and 3.5-9.

- The results of other laboratory and field studies indicate that at high cyclic stress ratios the relationship with $(N_1)_{60}$ is asymptotic and that very dense sands are strongly dilative, as shown on Figures 3.5-10 and 3.5-11.

2. Liquefaction of Filled Cavities. The stability of cavities has been conservatively analyzed for DBE level seismic shaking. The evaluation of the effects of 5.4g average spectral acceleration seismic shaking were found to be accommodated by the previous structural analysis and have negligible effect (well within range of the dynamic analysis) on the seismic response of structures as shown in Table 3.5-1.

3. Offshore Conduit Blockage. Conduit blockage was found not to be credible during or after 5.4g average spectral acceleration seismic shaking during minimum SWC operational conditions (i.e., one SWC pump per conduit). This is because the water velocity in the conduits would be high enough to prevent blockage of the conduit by erosion of sand material entering the conduit before the sand reached a depth of 11 to 14 feet as shown on Figure 3.5-12.

4. Stability of Cut Slopes. As shown on Figure 3.5-13, some permanent displacements within the terrace deposits could occur in cut slopes surrounding the plant area. Simplified relationships indicate that the 2:1 slopes to the east of the plant area (about 25 feet from the fuel handling building) could experience displacement on the order of 1/4 to 1-1/4 feet when subjected to the seismic shaking of 5.4g average spectral acceleration. The magnitude of the estimated movement would not produce adverse effects on the operation of this structure. The permanent displacements calculated for the 1/2:1 slopes to the northeast range from ½ to 3-1/4 feet. Considering the fact that these slopes are more than 140 feet from the closest critical facility, this range of displacement would have no effect on critical facilities.

TABLE 3.5-1
SUMMARY OF MAXIMUM EFFECTS OF CAVITIES ON STRUCTURES

Structure	Well No.	Maximum Decrease of Total Basemat Dynamic Stiffness* (percent)	
		Translation	Rocking
Containment Unit 3	8	4	5
Auxiliary Building	6, 7	2	2
Fuel Handling Building Unit 2	6	<1	3
Fuel Handling Building Unit 3	7,8	<1	8

3.6 ANALYSIS OF PLANT SYSTEMS AND SEQUENCES

This section describes the process and results used to quantify the seismic-induced core damage frequency for SONGS 2/3 and is organized into the following subsections:

3.6.1 Methodology
3.6.2 Seismic Event Tree
3.6.3 Input Information
3.6.4 Human Reliability Analysis (For Seismic Related Failures)
3.6.5 Nodal Equations
3.6.6 Quantification of Seismic Damage States
3.6.7 Non-Seismic Failures and Human Reliability Analysis (For Random Failures)
3.6.8 Overall Quantification Results
3.6.9 Sensitivity and Uncertainty Analysis

Tier 2 documents contain the detailed relay chatter evaluations, computer code output files, conditional core damage probability calculation files and sensitivity/uncertainty analysis files.

3.6.1 METHODOLOGY

The purpose of this portion of the analysis is to:

- delineate the potential seismic-induced structural and equipment failure scenarios that could occur after a seismic event,
- quantify the frequencies of these seismic damage scenarios,
- quantify the conditional core damage probabilities for these scenarios, including non-seismic failures and human interactions, and
- quantify the overall frequency of seismic-induced core damage.

Traditional event tree techniques were used to delineate the potential combinations of seismic-induced failures and resulting seismic scenarios which were termed "seismic damage states." The frequencies of these seismic damage states were quantified by convolving the SONGS site-specific mean earthquake hazard curve with the structure and equipment seismic fragility curves. This quantification included dependent and correlated failures and appropriate success states. For those scenarios that required additional non-seismic failures to occur to result in core damage, the IPE internal events model (event trees and fault trees) was used to develop conditional core damage probabilities (CCDP) with appropriate changes given the seismic damage state while incorporating random failures of equipment and operator actions. The overall seismic-induced core damage frequency (CDF) was then quantified from these intermediate results. The seismic risk quantification process is shown in Figure 3.6-1.

3.6.2 SEISMIC EVENT TREE

This section describes the development and structure of the seismic event trees.

3.6.2.1 Development of the Seismic Event Trees

The seismic event tree (SET) was developed to delineate the potential successes and failures that could occur due to a seismic event based on the structures/components and relays/switches listed in Tables 3.6-1. The SET is depicted in Figure 3.6-2 and the event headings and seismic damage states are discussed below.

The selection of event tree nodes and their order in the event trees was based on the following considerations. First, from a review of the fragilities, it was clear that the fragilities associated with loss of offsite power were much lower than other components and structures. Therefore, the first event tree node addresses whether offsite power remains available or is lost due to the seismic event.

The next node is a potential seismic-induced anticipated transient without scram event (ATWS). While all reactor scram equipment have high capacity, the relatively large uncertainty parameters for the reactor vessel internals caused this equipment to be slightly above the screening criterion of 1E-7/year. Some ATWS situations can be mitigated, depending on the moderator temperature coefficient, reactivity conditions in the fuel, and availability of emergency boration and heat removal. The ATWS node only includes seismic failure of reactor vessel internals with ATWS mitigation features included in a following event tree node.

The next step in SET development is different than internal events event tree development since support systems and structures are included as top events for the SET. Seismic events that lead directly to core damage are positioned toward the front of the SET to reduce the number of sequences that are needed. Those events that damage critical support systems are typical of direct seismic core damage sequences. Therefore, the next SET nodes are loss of instrumentation and control, loss of 125VDC buses, and loss of the emergency diesel generators.

The seismic-induced small Loss of Coolant Accident (LOCA) event tree node was initially placed before the emergency switchgear node since it had not been determined whether emergency switchgear/inverter room cooling success criteria would be impacted by the electrical loads required to mitigate a LOCA event. Subsequent analysis showed that the switchgear/inverter room cooling is not impacted by LOCA loads. The condensate storage tanks (CST) node, which is required for Auxiliary Feedwater (AFW) success, was placed before the Component Cooling Water (CCW) and Saltwater Cooling (SWC) since AFW success in a non-LOCA sequence would be sufficient to prevent core damage. The safety injection and emergency boration node is required for a seismic-induced LOCA or ATWS. CCW and SWC precedes the safety injection node since long term Emergency Core Cooling System (ECCS) requires CCW and SWC.

The last node on the SET is failure of the turbine-driven AFW pump. This node does not lead directly to core damage, since the motor-driven AFW pumps may be available. It is only addressed on sequences with success of the CST node.

Potential impacts of seismic-induced relay chatter are included in the SET nodal equations for the relays which did not meet the seismic capacity screening criteria. These relays are listed in Table 3.6-1. The operator actions to correct relay chatter are also included as appropriate. These actions are discussed more in Section 3.6.3.

Only the seismic-induced impacts are treated in the SET; therefore, success of equipment in the SET does not imply success from non-seismic causes of failure. Non-seismic failures, such as random failure of a pump or an operator error, are evaluated in the internal events model and are included in the overall quantification.

3.6.2.2 SET Top Events

The definitions of the top events in the SET are as follows:

Top Event	**Success/Failure Description**
S	Seismic event greater than 0.25g average spectral ground acceleration, which is approximately 0.1g peak ground acceleration.
OP	Offsite power remains available. Failure is the loss of power to the emergency 4kV buses (A04 and A06) from offsite power sources (which includes the reserve auxiliary transformer and switchyard).
IN	Instrumentation and control remains available. Success implies that the operators are able to control equipment from the main control room with instrumentation and controls available. Failure implies severe loss of instrumentation and control which is conservatively assumed to lead to core damage.
DC	DC power is available. Failure leads to loss of control power and instrumentation for many components, including the emergency diesel generators. The turbine-driven auxiliary feedwater pump (TDAFWP) may be available for secondary heat removal; however, without DC powered steam generator (SG) level indication, AFW was conservatively assumed to fail with consequential core damage.
DG	Emergency electric power is available to the equipment served by the emergency diesel generators (DG). Failure implies an immediate station blackout (SBO). The TDAFWP may be available for reactor coolant system (RCS) heat removal. However, eventual battery discharge and

loss of SG level instrumentation results in loss of AFW control with consequential core damage.

SL — No seismic-induced small LOCA occurs. Failure implies a small LOCA (less than 2" equivalent diameter) has occurred due to the seismic event. This could be caused by multiple failures in the small instrument lines connected to the RCS. The safety injection function is required to mitigate a seismic-induced small LOCA. (Larger LOCAs were screened based on strong seismic ruggedness.)

SWR — The emergency 4kV and 480V switchgear and MCCs are available. Failure results in a situation similar to station blackout with only battery power initially available. Although the TDAFWP may initially be available, it is conservatively assumed that loss of the batteries and SG level indication will result in core damage.

CST — CST121 and CST120 are available. Both tanks are required to supply water to the AFW pumps for long term decay heat removal; therefore, failure of either tank results in AFW failure and consequent core damage occurs. This is conservative since either tank can provide adequate feedwater for several hours before cooling down and depressurizing the primary system so that shutdown cooling can be initiated. However, this cooldown is conservatively not modeled.

CC — CCW and SWC are available. Failure of CCW and SWC results in failure of cooling for the safety injection pump lube oil coolers, shutdown heat exchangers, and the emergency chilled water (ECW) system. Although AFW may be available for some time, eventual loss of switchgear/inverter room cooling (which is dependent on ECW) results in loss of DC power. Loss of instrumentation, which depends on DC power, is assumed to result in consequential core damage.

RW — Refueling water storage tank (RWST) available for injection or emergency boration. Success implies post-seismic availability of the RWST to provide borated water for safety injection to mitigate a small LOCA or for emergency boration operations in the event of an ATWS. Failure implies loss of these functions as well as loss of containment sprays. There are different nodal equations depending whether it is associated with a SLOCA or ATWS sequence. In an ATWS, the nodal equation also includes the non-seismic failure associated with unfavorable moderator temperature coefficient and early cycle reactivity conditions.

TD Turbine driven AFW pump available. Failure implies that the TDAFWP is not available for decay heat removal. Loss of the TDAFWP does not lead directly to core damage since other non-seismic failures must occur.

3.6.3 INPUT INFORMATION

As discussed above, site-specific data was collected and incorporated into the Seismic Event Tree. This data includes the SONGS site-specific mean earthquake hazard curve and structural and equipment seismic fragility curves.

3.6.3.1 Seismic Hazard Curve

The methodology and results of the site-specific SONGS 2/3 seismic hazard curve development task were described in Section 3.1.1. Table 3.1-4 lists average spectral ground accelerations (0.5 -25 Hz) for the range of exceedance frequencies considered in this analysis.

As described in NUREG-1407, mean acceleration values were used in this analysis to identify any potential seismic vulnerabilities. Peak ground spectral accelerations (S_a), averaged over the spectral frequencies from 1 to 10 Hz, were used as the ground motion parameter since this parameter is judged to be more appropriate than the peak ground acceleration. The seismic hazard curve was quantified out to 8.0g S_a, which corresponds to an exceedance frequency of 1E-7/year. Further integration would not be significant to overall seismic core damage frequency and would not impact identification of dominant sequences, failure modes, or potential vulnerabilities.

3.6.3.2 Structure and Equipment Fragilities

The development of the seismic-induced failures that are incorporated into the quantification are discussed in Sections 3.4 and 3.5. Table 3.6-1, Seismic Structures and Equipment Fragilities, provides a summary of fragilities for those structures, equipment, relays and process switches that were not screened out as described below and included in the seismic model. As discussed in Section 3.2, plant systems and components providing safety functions to prevent core damage and safety functions of containment integrity, pressure suppression, heat removal, radioactivity removal, and isolation were included in this analysis.

Components and structures were screened from further analysis if their mean frequency of seismic failure was less than 1E-7/yr. This equates to a component or structural fragility of approximately 8.0g with uncertainty parameter values for β_u and β_r of 0.30. Most structures and components included on the walkdown list were screened out based on their high seismic capacity. Since any component or structure that contributes less than 1E-6/yr to the core damage frequency is unlikely to be considered a potential vulnerability, this criterion (1E-7/year) is considered to be conservative.

3.6.3.3 Relay and Process Switch Chatter Analysis

An extensive relay and process switch chatter analysis was performed for the SONGS 2/3 seismic IPEEE, with the following steps:

- Develop a comprehensive list of relays and process switches associated with equipment needed to mitigate post-seismic events and maintain hot shutdown.

- Perform a relay and process switch chatter evaluation to determine if chatter would be acceptable, if operator actions would be required, or if chatter is unacceptable.

- Determine the seismic capacity of the relays and process switches that were not chatter acceptable.

- Screen out relays and process switches whose capacity would result in a seismic failure frequency of less than 5E-7/year. The relay and process switch chatter criterion is higher than the structure/equipment criterion because chatter impacts are very likely to be mitigated by operator actions thus reducing the potential core damage frequency from a seismic damage state caused by chatter.

- Incorporate relays and process switches into nodal equations if they require operator action to mitigate their impact or are chatter unacceptable.

Over 1500 relays and 65 process switches were included in the chatter analysis task, with the result that most relays and switches were screened out based on chatter acceptability or high seismic capacity. These results and the chatter evaluation forms are contained in the Tier 2 documentation. Those relays and switches that were not screened out are listed in Table 3.6-1 and were included in the nodal equations in Section 3.6.5.

3.6.3.4 Seismic Dependencies

Seismic failures between similar redundant components (components in parallel) are conservatively considered completely dependent. Seismic failures among similar components which function in series are considered to be completely independent.

As with other components, those relays that are of the same type and located within a similar panel, with the same function are treated as totally correlated failures. That is, if the relay for train A fails during a seismic event, the relay for train B is also assumed to fail. This dependency is also modeled for shared systems between Units 2 and 3.

TABLE 3.6-1
SEISMIC STRUCTURES AND EQUIPMENT FRAGILITIES

Basic Event	Component Description	Component ID	S_A (g)	β_R	β_U
RATR1-2	Reserve Auxiliary Transformers	S21804ETXR1 S21804ETXR2	0.52	0.3	0.45
SWYD	Switchyard	---	0.74	0.2	0.34
PPMUTK	Primary Plant Makeup Tanks	S21203MT055 S21203MT056	3.03	0.2	0.2
SWCGATE	SWC Discharge Conduit Gate	---	4	0.2	0.2
SLOCA	Small LOCA	---	4	0.35	0.35
MCCS	Motor Control Centers	S21805ESBD S21805ESBE S21805ESBH S21805ESBJ S21805ESBQ S21805ESBRA S21805ESBRB S21805ESBS S21805ESBY S21805ESBZ	4.52	0.37	0.25
480SWGB04-6	480V Switchgear	S21805ESB04 S21805ESB06	5.1	0.3	0.28
AUXBLDG	Auxiliary Building	---	5.42	0.39	0.26
AFWGOV4700	TD AFW Pump Governor Valve	2SV4700	5.62	0.36	0.28
CCWHXE001-2	CCW Heat Exchangers	S21203ME001 S21203ME002	7.12	0.36	0.25
ECE335-6	Emergency Chiller Units	SA1513ME335 SA1513ME336	7.24	0.28	0.33
SEBLDG	Safety Equipment Building	---	7.5	0.37	0.25
RXINT	Reactor Internals	---	8	0.3	0.25
CST120	Condensate Storage Tank (Outer)	S21305MT120	8.44	0.35	0.25
CST121	Condensate Storage Tank (Inner)	S21305MT121	8.7	0.36	0.42

TABLE 3.6-1
SEISMIC STRUCTURES AND EQUIPMENT FRAGILITIES

Basic Event	Component Description	Component ID	S_A (g)	β_R	β_U
RWST005-6	Refueling Water Storage Tanks	S21204MT005 S21204MT006	8.7	0.36	0.42
RECIRC	Recirculation Sump Bellows	N/A	4.8	0.51	0.15
CCWBELL	CCW Pump Room Bellows	N/A	12.18	0.64	0.30
R112	DG Field Excitation	GE / HFA151	3.86	0.35	0.28
R132	DG Protection, Field Overexcitation	Basler/ BE-2-40	3.86	0.35	0.28
R133	DG Protection, Field Overexcitation	Eagle/HP-51-196	3.86	0.35	0.28
R122	DG Protection, Stator Ground	GE/IAV	6.79	0.30	0.28
R123	DG Protection, Loss of Excitation	West./CEH	6.79	0.30	0.28
R124	DG Protection, Volt Restraint O/C	GE / IJCV	6.79	0.30	0.28
R125	DG Protection, Volt Restraint O/C	GE / IJCV	6.79	0.30	0.28
R126	DG Protection, Volt Restraint O/C	GE / IJCV	6.79	0.30	0.28
R127	DG Protection, Reverse Power	GE / ICW	6.79	0.30	0.28
R128	DG Protection, Reverse Power	GE / ICW	6.79	0.30	0.28
R129	DG Protection, Reverse Power	GE / ICW	6.79	0.30	0.28
R135	DG Protection, Negative Phase	GE / INC	6.79	0.30	0.28
R21	CVCS Pump Start/Trip	GE / HFA51	4.39	0.34	0.23
R64	Battery Charger Protection, Hi Volt.	R10E3286-2	6.01	0.30	0.28
R81	Switchyard Protection	Various	0.74	0.20	0.34
R91	CCW Heat Exchanger Discharge	AGA / 7022	5.38	0.36	0.23
S11	DG Crankcase Pressure High Eng #1	SOR / 12NW66N4C1AJJTTX	3.8	0.37	0.35
S12	DG Crankcase Pressure High Eng #2	SOR / 12NW66N4C1AJJTTX	3.8	0.37	0.35

TABLE 3.6-1
SEISMIC STRUCTURES AND EQUIPMENT FRAGILITIES

Basic Event	Component Description	Component ID	S_A (g)	β_R	β_U
S13	DG Oil Pressure Low Eng #1	SOR / 4N6BB5NXC1AJJTTX12	3.8	0.37	0.35
S14	DG Oil Pressure Low Eng #2	SOR / 4N6BB5NXC1AJJTTX12	3.8	0.37	0.35
S15	DG Cooling Water Temp Hi Eng #1	SOR / 20XN6BB125JJTTX6	3.8	0.37	0.35
S16	DG Cooling Water Temp Hi Eng #1	SOR / 20XN6BB125JJTTX6	0.37	0.35	0.35

3.6.4 HUMAN RELIABILITY ANALYSIS (FOR SEISMIC RELATED FAILURES)

3.6.4.1 Methodology

The human reliability analysis (HRA) for the seismic IPEEE is largely based on the techniques, results and experience gained in completing the HRA for the SONGS 2/3 IPE [3-48]. The methodology relied largely on the Accident Sequence Evaluation Program (ASEP) methodology presented in NUREG/CR-4772 [3-49]. The HRA has two aspects: pre- and post-initiator operator actions. The pre-initiator operator actions, which are in the SONGS 2/3 IPE model, were not modified for the IPEEE. There are two types of post-operator actions: 1) actions which mitigate random, non-seismic failures and 2) actions that mitigate seismic-induced failures.

All operator actions were quantified using the ASEP method [3-49]. However, these operator actions (including actions in response to random, non-seismic failures) are impacted by seismic stress. To account for seismic effects on the operators, the operator action failure probabilities are modified by multiplying the operator failure probability with the appropriate seismic performance shaping factor (PSF).

The success of operator actions can be affected by seismic factors such as:

- severity of the seismic event,
- stress level of the operators,

- time required and time available to perform operator action, and
- location where operator actions are required.

A high severity seismic event may knock the operator to the ground, knock books off control room desks, and temporarily shock the operator. Lower severity events, such as seismic events where offsite power remains available, are not expected to affect the operators. Except for offsite power available sequences (where PSFs are set to 1.0), this analysis utilizes developed performance shaping factors (which assumes a high severity seismic event) given in Table 3.6-2.

The stress of the operator is assessed within the time/stress model of the ASEP calculation and not addressed within the seismic PSF.

Operator actions performed in the short-term following the seismic event are much more likely to be impacted. The decisions and execution of operator actions in the short-term must compete with other potentially distracting factors such as injuries to other personnel, confusion, shock, concerns for family, etc. Therefore, the PSFs are weighted higher for the first twenty minutes following the event. After the first twenty minutes and within the first hour, these factors should be reduced but will still have a significant residual impact. After an hour, the initial shock factors will have subsided, other staff are likely to be available to provide assistance, decision-making and execution of operator actions are more likely to be completed properly.

The location of where the operator must complete the required recovery actions may impact the likelihood of success. That is, although the operator is comfortable outside of the control room and knowledgeable of the plant, the majority of the operator's time is spent working in the familiar surroundings of the control room. After a seismic event, the operator is likely to be more oriented and in control of the plant status when in the control room than if he were out in the plant. In the control room, operator actions consist of manipulating switches and controls. Damage to the control room is likely to be less severe than in the plant. In the plant, the operator is expected to manually manipulate equipment near areas where non-seismically qualified equipment may have failed and fallen making it more difficult for the operator to complete the required actions. To account for the location of operator actions, the PSF for ex-control room actions are significantly greater than in-control room actions.

TABLE 3.6-2
SEISMIC PERFORMANCE SHAPING FACTORS

Seismic IPEEE Performance Shaping Factors	short time period (t < 20 min)	medium time period (20 m < t < 60 m)	long time period (1 hr < t < 24 hr)
In control room action	10	5	1
Ex-control room action	30	10	5

3.6.4.2 Recovery Actions for Seismic Related Failures and Relay Chatter Effects

Based on the existing procedures and discussions with simulator and training staff, viable recovery actions were identified for all non-screened relays and process switches. However, operator actions for the recovery of the effects of chattering of relays associated with offsite power were conservatively excluded from the analysis (i.e., non-recoverable). For all other non-screened relays and process switches, three operator actions were analyzed and included in the nodal equations.

In addition to the operator actions required to mitigate the effects of relay and process switch chatter, operator actions for the recovery of the effects of component seismic integrity failures were also analyzed. Three operator recovery actions were identified and analyzed in association with the seismic failure of the following components: a) primary plant makeup tank which provides makeup to the component cooling water system; b) emergency chiller units which provide cooling to the emergency switchgear and distribution rooms; and c) saltwater cooling discharge conduit heat treat gate whose failure could block return flow out to the ultimate heat sink.

Table 3.6-3, Seismic Operator Actions, summarizes the operator recovery actions, the time available for the action, the probability of operator action failure and the error factor associated with that probability. These operator actions were included in the nodal equations discussed in Section 3.6.5.

3.6.4.3 Recovery Actions for Random, Non-seismic Failures

Post-initiator operator actions for random, non-seismic failures are addressed in Section 3.6.7.

TABLE 3.6-3
SEISMIC OPERATOR ACTIONS

Basic Event	Description	Time Available	Initial Prob.	PSF	Final Prob	Error Factor
OP13	Operator fails to reset and start the diesel generator following relay or process switch chatter	55 Min.	1E-4	5	5E-4	10
OP64	Operator fails to reset battery charger relay	90 Min.	7E-3	5	3.5E-2	5
OP91	Operator fails to open SWC Pump discharge valve & start the redundant SWC pump	> 90 Min.	3E-4	5	1.5E-3	5
OPFIRETNKR	Operator fails to align fire truck for CCW makeup given PPMU tank is unavailable	4 Hr	3E-4	5	1.5E-3	5
OPALTVENT	Operator fails to respond to hi temp alarm in the switchgear/distribution room	4 Hr	3E-4	5	5.0E-2 (see note 1)	5
OPEMERLINE	Operator fails to open SWC Emergency Discharge Line to Seawall given gate failure	90 Min.	--	--	1.0E-1 (see note 2)	3

Note 1: An HRA value of 1.5E-3 was formally calculated. However, 5E-2 was conservatively used.
Note 2: Conservative screening value initially used. Refined calculation was not required.

3.6.5 NODAL EQUATIONS

Boolean equations were developed for each of the SET nodes based on the logic and seismic fragility information discussed in Section 3.6.2. The seismic fragilities table, Table 3.6-1, and the seismic operator actions table, Table 3.6-3, provide a cross-reference between the abbreviations used in the equations, the structure/component and operator action descriptions, and fragility or probability information. The failure and success equations for each top event are shown in Table 3.6-4.

An "&" (or "C" for operator actions and non-seismic events) in front of an event denotes success of the event. These equations, which represent the seismic failure or success of structures and components, are then combined into the seismic sequence equations as delineated by the SET. Both failures and successes are included in these seismic sequence equations. Each seismic sequence equation represents the Boolean logic associated with its corresponding seismic damage state (SDS). The actual sequence equations are listed in the Tier 2 documentation code output files.

3.6.6 QUANTIFICATION OF SEISMIC DAMAGE STATES

The seismic hazard curve information, structural/component fragilities, and SDS equations were then input to the EQESRA code to quantify the frequencies of the SDSs. The EQESRA code uses a discrete probability distribution (DPD) sampling process at each seismic magnitude interval to combine the seismic hazard frequency information with the seismic fragility information for each structure, component, relay and switch in the SDS equation. Successes, failures, and Boolean intersects are properly treated in this calculation. The code repeats this process for each seismic magnitude and then sums the results to obtain the SDS frequency. This process is then repeated for each SDS equation until all equations are quantified. The concepts and algorithms used in the EQESRA code are documented in the User's Manual [3-50]. The results files are contained in the Tier 2 documentation.

The results of the SDS quantification are summarized in Table 3.6-5, SONGS 2/3 Seismic Damage States.

As can be seen from Table 3.6-5, the only SDS for offsite power available that is significant is the first sequence which represents no seismic failures due to the earthquake. Essentially, this is a plant trip with offsite power and all safety systems available. For the loss of offsite power SDSs, only six have frequencies greater than 1E-6/yr. They are discussed in more detail in Section 3.6.8, Overall Quantification Results. The quantification of non-seismic failures in Section 3.6.7 utilize these SDS frequencies as initiating event frequencies, including the seismic failures as house events or guaranteed failures.

TABLE 3.6-4
SEISMIC EVENT TREE NODAL EQUATIONS

SET Node	Nodal ID Failure	Nodal ID Success	Equation
Seismic Event			No equation needed since this is the seismic event
Offsite Power	LOP		SWYD + RATR1-2 + R81
		OP	&SWYD * &RATR1-2 * &R81
Automatic Reactor Trip	AT		RXINT
		NAT	&RXINT
Instruments and Controls	IN		AUXBLDG + (OPALTVENT * ECE335-6)
		NI	&AUXBLDG * (COPALTVENT + &ECE335-6)
125VDC Buses	DC		R64 * OP64
		ND	&R64 + COP64
Diesel Generators	DG		(OP13 * (R112 + R132 + R133 + R122 + R123 + R124 + R125 + R126 + R127 + R128 + R129 + R135 + S11 + S12 + S13 + S14 + S15 + S16))
		NG	(COP13 + (&R112 * &R132 * &R133 * &R122 * &R123 * &R124 * &R125 * &R126 * &R127 * &R128 * &R129 * &R135 * &S11 * &S12 * &S13 * &S14 * &S15 * &S16))
Seismic-Induced Small LOCA	SL		SLOCA
		NL	&SLOCA
Emergency Switchgear	SWR		480SWGB04-6 + MCCS
		NSW	&480SWGB04-6 * &MCCS
Condensate Storage Tanks	CS1		CST121 + CST120
		NF1	&CST121 * &CST120
CCW and SWC (Small LOCA events Only)	CC		SEBLDG + CCWHXE001-2 + (R91 * OP91) + (PPMUTK * OPFIRETNKR) + (SWCGATE * OPEMERLINE) + CCWBELL
		NC	&SEBLDG * &CCWHXE001-2 * (&R91 + COP91) * (&PPMUTK + COPFIRETNKR) * (&SWCGATE + COPEMERLINE) * &CCWBELL
CCW and SWC (non-Small LOCA events	CCS		SEBLDG + (OPALTVENT * (CCWHXE001-2 + (R91 * OP91) + (PPMUTK * OPFIRETNKR) + (SWCGATE * OPEMERLINE) + CCWBELL))
		NCS	&SEBLDG *(COPALTVENT + (&CCWHXE001-2 * (&R91 + COP91) * (&PPMUTK + COPFIRETNKR) * (&SWCGATE + COPEMERLINE) * &CCWBELL))
Safety Injection (Non-ATWS)	RW		RWST005-6 + RECIRC
		NW	&RWST005-6 * &RECIRC
Emerg. Boration (ATWS Events)	EB		UNFAVMTC + RWST005-6 + R21
		NEB	CUNFAVMTC * &RWST005-6 * &R21
TD AFW Pump	TD		AFWGOV4700
		NT	&AFWGOV4700

TABLE 3.6-5
SONGS 2/3 SEISMIC DAMAGE STATES

SDS	Seismic Damage State Description	SDS Frequency
1	Offsite Power Available, No seismic failures of components	6.9E-3
2	Offsite Power Available, Seismic failure of TDAFW Pump	4.0E-12
3	Offsite Power Available, Seismic failure of CCW/SWC	1.7E-12
4	Offsite Power Available, Seismic failure of Condensate Storage Tanks	2.0E-12
5	Offsite Power Available, Seismic Failure of Emergency Switchgear	1.2E-10
6	Offsite Power Available, Seismic-Induced Small LOCA	7.0E-9
7	Offsite Power Available, Small LOCA, Seismic failure of TDAFWP	5.0E-17
8	Offsite Power Available, Small LOCA, Seismic failure of RWST	3.0E-12
9	Offsite Power Available, Small LOCA, Seismic failure of CCW/SWC	4.6E-14
10	Offsite Power Available. Small LOCA, Seismic failure of Condensate Tanks	4.7E-14
11	Offsite Power Available, Small LOCA, Seismic failure of Emergency Switchgear	1.5E-12
12	Offsite Power Available, Small LOCA, Seismic failure of DC Power	1.8E-13
13	Offsite Power Available, Small LOCA, Seismic failure of Instruments/Control	5.3E-11
14	Offsite Power Available, ATWS Event	4.0E-17
15	Offsite Power Available, ATWS Event, Seismic failure of Emergency Boration	4.7E-18
16	Loss of Offsite Power, No seismic failure of other components	6.9E-3
17	Loss of Offsite Power, Seismic failure of TDAFWP	1.3E-6
18	Loss of Offsite Power, Seismic failure of CCW/SWC	5.4E-7
19	Loss of Offsite Power, Seismic failure of Condensate Storage Tanks	4.6E-7
20	Loss of Offsite Power, Seismic Failure of Emergency Switchgear	6.6E-6
21	Loss of Offsite Power, Seismic-Induced Small LOCA	1.5E-5
22	Loss of Offsite Power, Small LOCA, Seismic failure of TDAFWP	1.9E-7
23	Loss of Offsite Power, Small LOCA, Seismic failure of Injection/Recirculation	8.0E-7
24	Loss of Offsite Power, Small LOCA, Seismic failure of CCW/SWC	1.9E-7
25	Loss of Offsite Power, Small LOCA, Seismic failure of Condensate Tanks	9.3E-8
26	Loss of Offsite Power, Small LOCA, Seismic failure of Emergency Switchgear	1.2E-6
27	Loss of Offsite Power, Seismic failure of DGs	7.6E-8
28	Loss of Offsite Power, Seismic failure of DC Power	3.0E-8
29	Loss of Offsite Power, Seismic failure of Instruments/Control	3.3E-6
30	Loss of Offsite Power, ATWS Event	6.2E-8
31	Loss of Offsite Power, ATWS Event, Seismic failure of Emergency Boration	4.8E-8

3.6.7 NON-SEISMIC FAILURES AND HUMAN RELIABILITY ANALYSIS (FOR RANDOM FAILURES)

For those seismic damage states (SDS) with frequency greater than 1E-7/year, the impact on the plant and plant systems was evaluated using the internal events IPE models, but modified to reflect the special conditions for a seismic event. As shown in Table 3.6-5, only 12 SDSs (1, 16, 17, 18, 19, 20, 21, 22, 23, 24, 26, and 29) are greater than 1E-7/year.

The only SDS from the offsite power available portion of the SET that could not be screened is the first SDS with seismic success of all safety systems (SDS #1). The calculation of the conditional core damage frequency for SDS #1, which accounts for additional non-seismic random failures, was treated similarly to the Loss of Power Conversion System initiating event from the internal events IPE analysis. That is, only the MFW/condenser systems are modeled inoperable following plant trip.

Of the remaining 11 SDSs, seven sequences (SDS 18, 19, 20, 23, 24, 26, and 29) directly result in core damage. Although AFW may be available for several hours for some of these sequences, it is assumed to fail after loss of instrumentation. Therefore, no conditional core damage probability calculation of non-seismic, random failures is required. As presented in Table 3.6-7, the conditional core damage probability (CCDP), given the seismic failures is 1.0 or guaranteed failure.

The internal events IPE models were used to determine CCDPs for the remaining five SDSs (1, 16, 17, 21, 22). In the loss of offsite power SDSs, the IPE models were modified to reflect the seismic failures and the assumed loss of offsite power for 24 hours. The diesel generator mission time was increased to 24 hours and the offsite power recovery was set to 0.0 (no offsite power available).

Those operator actions from the IPE that are found to mitigate the impact of seismic-induced loss of offsite power or small break LOCA are modeled and modified by use of the seismic performance shaping factors discussed in Section 3.6.4.1. Table 3.6.6, Non-Seismic Operator Actions, represents changes to the internal events IPE model.

The REBECA PRA software, with these modifications, was used to calculate CCDPs. The output files (including cut sets) and associated event trees are contained in the Tier 2 documentation.

TABLE 3.6-6
NON-SEISMIC OPERATOR ACTIONS

Basic Event	Description	IPE Prob (EF)	In/ Out	T	PSF	IPEEE Prob (EF)
D-HCAUXSPRYU	Operator fails to align charging to auxiliary pressurizer spray	6E-4 (5)	see note 1			Not applicable (N/A)
D-HCBORATE-U	Operator fails to emergency borate in an ATWS event	3E-3 (10)	see note 2			N/A
D-HCNOSIAS2U	Operator fails to respond to high temp alarm to charging pp rooms given no SIAS	5E-3 (10)	Seismic specific HRA performed			0.05
E-HCP024—U	Operator fails to attempt to start standby CCW pp P024	1E-3 (10)	IN	30min	5	5E-3 (10)
F-HCDEPRESSU	Operator fails to depressurize SG's w/i 1 hr & align condensate	4E-2 (5)	see note 3			1.0 [4E-2 (5)]
FGHCCNDREC-U	Operator fails to recover condensate after loss of PCS w/i 60 min	6E-2 (3)	see note 3			1.0 [6E-2 (3)]
FGHCMFWREC-U	Operator fails to recover MFW after loss of PCS	4.3E-1 (3)	see note 3			1.0 [4.3E-1 (3)]
H-HC9300&01U	Operator fails to close HV-9300 and 9301 per SO23-12-3	5E-3 (5)	IN	>1 HR	1	5E-3 (5)
H-HCHLRECRCU	Operator fails to establish hot leg recirculation w/i 4 hrs	5E-5 (5)	IN	>1 HR	1	5E-5 (5)
K-HCSCRAM--U	Operator fails to manually SCRAM reactor	3E-3 (10)	note 2			N/A
L-HCBYPASS-V	Operator fails to open HV-4762 or 4763 w/i 1 hr w/ procedure	4E-4 (3)	IN	1 HR	1	4E-4 (3)
L-HCCRCNNCTU	Operator fails to cross-connect MDAFWP to opposite SG	7E-3 (5)	IN	1 HR	1	7E-3 (5)
L-HCCSTMU--U	Operator fails to provide CST makeup per proc.	2E-5 (5)	OUT	8 HR	5	1E-4 (5)
L-HCNSBOMANU	Operator fails to manually control AFW flow valves w/i 2.5 - 8 hrs	6E-3 (10)	OUT	5.5 HR	5	3E-2 (10)
L-HCSBO-MANU	Operator fails to manually operate the TDAFWP w/o DC power w/i 1 hr (SBO event)	6E-3 (5)	OUT	2.5 HR	5	3E-2 (5)
L-HCTP140--U	Operator fails to manually operate TDAFW pump (no DC power @ 8 hr; non SBO)	6E-3 (5)	OUT	1 HR	5	3E-2 (10)
L-HCTP1401HU	Operator fails to manually open HV-4716 (stop valve) w/i 1 hr given battery 9 fails	1.2E-2 (10)	calculated same as L-HCTP140-U			3E-2 (10)
M-HCE-331--U	Operator fails to align chiller E331 to backup E330	5E-2 ([illegible]	OUT	1.5 HR	5	0.25 (10)

TABLE 3.6-6
NON-SEISMIC OPERATOR ACTIONS

Basic Event	Description	IPE Prob (EF)	In/ Out	T	PSF	IPEEE Prob (EF)
M-HCNOSIAS-U	Operator fails to respond to hi temp alarm in the switchgear/distribution room given no SIAS	3.3E-4 (10)	IN	3 HR	1	3.3E-4 (10)
M-HCGASFANS	Operator fails to cool electrical rooms with gas fans (Note: this action is mutually exculsive of M-HCNOSIAS-U)	3E-4 (5) (conserv used 5E-2)	OUT	3 HR	5	1.5E-3 (5) (conserv used 5E-2)
MRHC3E335--U	Operator fails to cross-tie CCW to chiller E335 from other unit given SGTR	3E-3 (5)	see note 1			N/A
MRHC3E336--U	Operator fails to cross-tie CCW to chiller E336 from other unit given SGTR	3E-3 (5)	see note 1			N/A
N-HCCSINJCTU	Operator fails to depressurize below CS pump shutoff head given HPSI failure (Induced LOCA)	0.1 (10)	IN	≈ 1 HR	5	0.5
NBHCCSINJCTU	Operator fails to depressurize below CS pump shutoff head given HPSI failure (Medium LOCA)	1	no factor applied			1.0
NCHCCSINJCTU	Operator fails to depressurize below CS pump shutoff head given HPSI failure (Induced LOCA)	0.5 (10)	IN	1 HR	1	0.5
T-HCDEPRESEU	Operator fails to depressurize RCS early (SGTR)	1E-3 (5)	see not e 1			N/A
T-HCINDLOCAU	Operator fails to depressurize RCS & control ECCS flow - induced LOCA	7.5E-2 (10)	IN	1 HR	5	0.4 (10)
TDHCCSSPRAYU	Operator fails to depressurize & cooldown RCS to reduce RCS leak (SSL)	6E-3 (10)	IN	> 1 HR	1	6E-3 (10)
TRHCADV-P---U	Operator fails to manually operate ADV locally given SGTR	3E-3 (10)	see note 1			N/A
U-HCSHED60MU	Operator fails to loadshed within 1 hour	6E-3 (5)	IN	1 HR	1	6E-3 (5)

Note 1: Basic event is used in the SGTR analysis and is not applicable to the seismic IPEEE.

Note 2: Basic event is used in the ATWS analysis which has been screened out (<1E-7/yr).

Note 3: Basic event is only used if offsite power is available. The severity of the seismic event where offsite power remains available is likely to be small. The operators stress level should not be very high since offsite power and all safety systems are likely to be available. Therefore, if offsite power is available, the PSF = 1.0. If offsite power is unavailable, set the probability to 1.0.

3.6.8 OVERALL QUANTIFICATION RESULTS

To obtain the overall results, the frequency of each seismic damage state (SDS) is multiplied by the conditional core damage probability (CCDP) for that SDS. (Note that those seismic damage states less than 1E-7/year were previously screened out and are not included here.) Since the frequency of each SDS is independent of the others, the total core damage frequency due to seismic events, 1.7E-5/year, is simply the summation of the individual SDS sequence core damage frequencies. The results are given in Table 3.6-7 and discussed below.

The highest ranked sequence, Sequence 1, represents a seismic-induced loss of offsite power and seismic loss of emergency switchgear event with a sequence CCDP of 1.0. The core damage frequency for this sequence is dominated by the seismic failure of the motor control centers. The loss of the chargers powered by the 480V MCCs will eventually fail DC control power and indication which is assumed to result in core damage.

Sequence 2 represents a seismic-induced loss of offsite power with seismic failure of instrumentation and control. This seismic sequence requires no additional random failures (CCDP=1.0) and is dominated by the structural failure of the auxiliary building, which contains the control room, resulting directly in core damage.

Sequence 3 characterizes a seismic-induced loss of offsite power with no seismic failures of other components. Seismic failure of offsite power is governed by the failure of the transformer fluid cooling fins on the reserve auxiliary transformers. Relay chatter associated with offsite power was also modeled with operator recovery actions conservatively not modeled. The CCDP for this sequence is 3.57E-4 with dominant cutsets involving random failures of the diesel generators or associated support systems such as room ventilation resulting in a station blackout event.

Like Sequence 3, Sequence 4 represents a seismic-induced loss of offsite power with no seismic failures of other components. The CCDP for this sequence is 1.89E-4 with dominant cutsets involving random failures of the condensate makeup system and the turbine-driven and motor-driven AFW pumps.

Sequence 5 depicts a seismic-induced loss of offsite power and small LOCA with seismic loss of emergency switchgear event with a sequence CCDP of 1.0. As discussed for Sequence 1 above, the core damage frequency for this sequence is dominated by the seismic failure of the motor control centers.

These sequences, Sequences 1-5, contribute over 85% to the total seismic core damage frequency. Refer to Table 3.6-7 for descriptions of the dominant seismic and random failures for the remaining 12 sequences.

The contribution of seismic spectral acceleration ranges to the overall core damage frequency was calculated and is shown in Figure 3.6-3, Seismic Event Magnitude vs. Core Damage Frequency. This calculation was performed considering both seismic and random, non-seismic failures. As shown in Figure 3.6-3, approximately 50% of the overall core damage frequency was attributed to seismic initiating events with spectral accelerations 2.0g or less. These seismic events have relatively high frequencies of occurrence when compared to those events in the higher acceleration ranges. Results of the plant level fragilities analysis is presented in Figure 3.6-4, "Plant Level Fragility Curves." As shown in this figure, the plant HCLPF[3] is approximately the same as the SONGS 2/3 safe shutdown earthquake (SSE) (0.67 pga, or approximately 1.5 gSA). The plant median capacity (the acceleration corresponding to the median curve with 50% conditional probability of failure) is about 2.5 times the SSE.

3 HCLPF is defined as the acceleration corresponding to the 95% confidence bound that the conditional probability of failure is 5% or less.

TABLE 3.6-7
SONGS 2/3 TOP SEISMIC CORE DAMAGE SEQUENCES

Seq. Rank #	SDS	Sequence Description	Seismic Failures	Dominant Random Failures	SDS Frequency	CCDP	Sequence CDF
1	20	Seismic Loss of Offsite Power Seismic Failure of Emergency Switchgear	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) 480V Switchgear Motor Control Centers	N/A	6.57E-6	1.0	6.57E-6
2	29	Seismic Loss of Offsite Power Seismic Failure of Instruments/Control	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Auxiliary Building Emergency Chillers	N/A	3.31E-6	1.0	3.31E-6
3	16	Seismic Loss of Offsite Power No Seismic Failure of Other Components Random Loss of Emergency Diesel Generators	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter)	Emergency Diesel Generators DG Emergency Supply Fans DG Fuel Transfer Pumps	6.94E-3	3.57E-4	2.48E-6
4	16	Seismic Loss of Offsite Power No Seismic Failure of Other Components Random Loss of AFW	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter)	Operator Failure - Condensate M/U Turbine Driven AFW Pump Motor Driven AFW Pumps Operator Failure - Battery Chrgr Failure TD AFW Pump Control Valves Emergency Chillers	6.94E-3	1.89E-4	1.31E-6
5	26	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of Emergency Switchgear	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA 480V Switchgear Motor Control Centers	N/A	1.15E-6	1.0	1.15E-6
6	23	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of Recirculation	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA Emergency Sump Valve Bellows	N/A	8.01E-7	1.0	8.01E-7

TABLE 3.6-7
SONGS 2/3 TOP SEISMIC CORE DAMAGE SEQUENCES

Seq. Rank #	SDS	Sequence Description	Seismic Failures	Dominant Random Failures	SDS Frequency	CCDP	Sequence CDF
7	18	Seismic Loss of Offsite Power Seismic Failure of CCW/SWC	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Safety Equipment Building CCW Heat Exchangers SWC Valve Relays (Chatter) Primary Plant M/U Tank SWC Discharge Gate	N/A	5.37E-7	1.0	5.37E-7
8	19	Seismic Loss of Offsite Power Seismic Failure of Condensate Storage Tanks	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Condensate Storage Tank T120 Condensate Storage Tank T121	N/A	4.56E-7	1.0	4.56E-7
9	21	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Random Loss of CCW Random Loss of Safety Injection	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA	CCW Heat Exchangers CCW Non-Critical Loop Isolation Vlvs CCW Pumps High Pressure Safety Injection Pumps	1.49E-5	2.13E-2	3.18E-7
10	24	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of CCW/SWC	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA Safety Equipment Building CCW Heat Exchangers SWC Valve Relays (Chatter) Primary Plant M/U Tank SWC Discharge Gate	N/A	1.87E-7	1.0	1.87E-7
11	21	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Random Loss of Recirculation	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA	High Pressure Safety Injection Trains Emergency Sump Recirculation Valves RWST Outlet Check Valves	1.49E-5	3.10E-3	4.62E-8

TABLE 3.6-7
SONGS 2/3 TOP SEISMIC CORE DAMAGE SEQUENCES

Seq. Rank #	SDS	Sequence Description	Seismic Failures	Dominant Random Failures	SDS Frequency	CCDP	Sequence CDF
12	1	Offsite Power Available No seismic Failures of components Random Loss of MFW/Condensate	N/A	Non-recovery of Main Feedwater Non-recovery of Condensate Operator Fails to provide CST M/U Turbine Driven AFW Pump Motor Driven AFW Pumps	6.93E-3	2.01E-6	1.39E-8
13	1	Offsite Power Available No Seismic Failures of Components Random Failure of PZR Safety Valves to Reclose Random Loss of Recirculation	N/A	Pressurizer Safety Valves Emergency Sump Recirculation Valves	6.93E-3	7.23E-7	5.01E-9
14	22	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Seismic Failure of TDAFWP Random Loss of Safety Injection	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA TDAFWP Governor Valve	Emergency Diesel Generators CCW Heat Exchangers CCW Non-Critical Loop Isolation Vlvs	1.87E-7	1.68E-2	3.15E-9
15	1	Offsite Power Available No Seismic Failures of Components Random Failure of PZR Safety Valves to Reclose Random Loss of Safety Injection	N/A	Pressurizer Safety Valves Operator Fails to Align Cs for Injection High Pressure Safety Injection Pumps CCW Heat Exchangers	6.93E-3	4.29E-7	2.97E-9
16	21	Seismic Loss of Offsite Power Seismic-Induced Small LOCA Random Loss of Condensate Random Loss of Auxiliary Feedwater Pumps	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) Small LOCA	Operator Fails to Provide Cst M/u AFW Turbine Driven Pump Motor Driven AFW Pump Emergency Diesel Generator	1.49E-5	1.00E-4	1.49E-9
17	17	Seismic Loss of Offsite Power Seismic Failure of TDAFWP Random Loss of Auxiliary Feedwater	Switchyard Reserve Auxiliary Transformers Switchyard Relays (Chatter) TDAFWP Governor Valve	Motor Driven AFW Pumps	1.27E-6	1.06E-3	1.35E-9

3.6.9 SENSITIVITY AND UNCERTAINTY ANALYSIS

Several sensitivity studies and uncertainty analysis were performed to examine different input information and assumptions.

3.6.9.1 Sensitivity Studies

During the development of the SET, modeling of system component failures and successes for ATWS sequences was simplified to exclude those component failures and successes (nodes) which have an insignificant impact on the calculated core damage frequency. As shown in Figure 3.6-2, Seismic Event Tree, only the success and failure of the injection and emergency boration node was considered for an ATWS failure. A sensitivity study was performed to characterize the contribution of those excluded nodes. Fifteen (15) sequences or SDS resulted from this study using the following nodes in addition to the ATWS failure nodal equation: instruments and controls, 125VDC buses, diesel generators, seismic-induced LOCA, condensate storage tanks, CCW and SWC, injection and emergency boration, and turbine driven AFW pump. The results of this sensitivity study confirmed the assumption that the contribution of the excluded nodes was insignificant. The combined contribution of all 15 sequences was less than 1E-7/yr.

Another sensitivity study was performed to address the impact of unacceptable relay chatter associated with the suction and discharge valves on the CCW swing pump. Because this pump is a third-of-a-kind pump within the CCW system, this failure was analyzed separately to determine whether the resulting SDS frequency would exceed the screening criterion for inclusion in the SET. Using the fragility values for the relays, the CCW swing pump frequency of failure was determined. The internal events IPE models were used to calculate the CCDP assuming CCW swing pump failure. The resulting seismic core damage frequency was less than 1E-7/yr. Therefore, the chatter of these relays was considered to have minimal risk significance and was not included in the SET.

As discussed in Section 3.6.4.1, six seismic operator actions were analyzed and quantified as part of the SET. Sensitivity studies for each operator action were performed by setting each operator action failure probability to 1.0. Only one operator action sensitivity study resulted in an increase of any significance. This study addressed the impact of the operator failing to reset and start the diesel generator following relay or process switch chatter. The seismic core damage frequency increased an order of magnitude due to the effects of non-recovery of DG relay and process switch chatter. The remaining 5 sensitivity studies showed that changing the operator action failure probabilities to 1.0 increased the seismic core damage frequency less than 15%.

3.6.9.2 Uncertainty Analysis

Statistical and/or modeling uncertainty in the seismic CDF results can come from the hazard curve uncertainty, the fragilities uncertainties, and non-seismic uncertainties in the CCDP calculation. While a complete hazard uncertainty analysis was performed as discussed in Section 3.1.1, the quantification of the seismic core damage frequency included only the mean hazard curve. Although the performance of a quantitative hazard uncertainty analysis is not required by the NRC, uncertainty analysis was performed using the 85th percentile hazard curve. The dominant seismic sequences did not change; however, the ranking of two sequences exchanged positions. The seismic damage state frequencies for these two sequences increased, however the relative impact of this increase for one of these sequences was reduced by its associated CCDP, which did not change. The frequencies of four additional seismic damage states were raised slightly above the screening criteria of 1E-7/yr. These are not considered significant; therefore, no additional vulnerabilities were identified. The overall seismic core damage frequency increased less than a factor of two when the 85th percentile hazard curve was utilized.

The EQESRA code quantification included the fragilities uncertainties, expressed by the random and modeling uncertainty parameters (β_R, β_U) given in Table 3.6-1. Statistical uncertainties in the CCDP calculations were not modeled for this analysis, but based on the internal events IPE, would be about a factor of 3 to 5 for the 95 percent confidence level.

Based on the above discussion, the uncertainty analysis did not significantly change or alter the qualitative results or the identification of dominant sequences, contributors, or vulnerabilities.

3.7 ANALYSIS OF CONTAINMENT PERFORMANCE

Generic Letter 88-20, Supplement 4, Appendix 2 states:

"The evaluation of the containment performance (Level II analysis) for external events should be directed toward a systematic examination of whether there are sequences that involve containment failure modes distinctly different from those identified in the

IPE internal events evaluation or contribute significantly to the likelihood of functional failure of the containment (i.e., loss of containment barrier independent of core melt)."

In Appendix 2 of Generic Letter 88-20, Supplement 4, the following aspects of this assessment are listed:

- Identify mechanisms that could lead to containment bypass.
- Identify mechanisms that could cause failure of the containment to isolate.
- Determine the availability and performance of the containment systems under the external hazard to see if they are different from those evaluated under the internal event hazard evaluation.

NUREG-1407 also requested specific items to be evaluated for seismic vulnerabilities:

- gross structural failure of the containment (section 3.7.3),
- failure of major equipment or structures inside containment at very high accelerations (section 3.7.3),
- isolation failure due to relay chatter (section 3.7.2),
- walkdown of containment penetrations for spatial interactions and unique configurations (section 3.7.2),
- hatch inflatable seals and associated air system (section 3.7.2),
- penetration cooling (section 3.7.2),
- containment isolation actuation systems (section 3.7.2),
- backup air system for isolation AOVs (section 3.7.2), and
- components of containment heat removal/pressure suppression system such as fan coolers, support systems, and system interaction effects (section 3.7.3).

Each of the above items was appropriately included in the evaluation of seismic capacity and containment performance.

The plant walkdowns for the seismic IPEEE (discussed in Section 3.3) did not result in the identification of any additional or unique seismic-related containment failure modes. The seismic events are therefore modeled in the same manner as an internal event with regard to random failures, phenomenological response, and containment response, but include the equipment losses due to the seismic event.

The potential impact of seismic events on containment bypass is addressed in Section 3.7.1. The potential impact of seismic events on containment isolation is addressed in Section 3.7.2. The results of an evaluation of the potential seismic event impact on containment systems are discussed in Section 3.7.3, and the potential for release of radioactivity from the plant in the event of a seismic event is discussed in Section 3.7.4.

3.7.1 CONTAINMENT BYPASS POTENTIAL

The potential for seismic-induced interfacing systems LOCAs (ISLOCA) involves the failure of the RCS pressure boundary leading to a LOCA outside the containment boundary. The internal events IPE has identified all potential ISLOCA paths and was used as the initial basis for this seismic analysis. Valves in each of the ISLOCA paths were reviewed for inclusion on the seismic equipment list (SEL) and then included in the seismic capacity walkdown. Paths with check valves and normally closed manual valves for isolation have high seismic capacity. These paths were not evaluated further. Power operated valves, such as MOVs and AOVs, were included in the SEL and walkdown. The relays associated with these valves, including isolation actuation systems, were also included in the relay chatter evaluation. Particular attention was placed on the shutdown cooling lines and valves and the CVCS letdown lines. Based on the ISLOCA evaluation, there are no seismic vulnerabilities associated with these paths, valves and associated relays. No additional containment performance modeling is necessary.

The other bypass potential evaluated for the seismic IPEEE is a seismic-induced steam generator tube rupture (SGTR). A seismic event will not directly cause a steam generator tube to rupture based on the high capacity of the steam generator and its internals (median capacity > 8g). All core damage sequences, however initiated, that progress with a high primary system pressure have a small probability to "induce" a SGTR. The combined effects of very high temperatures from the melting core off-gases and high differential pressures may cause failure of a tube. This is not unique to seismic event-initiated sequences. Therefore, the dependent probabilities are the same as for internal event-initiated sequences. These induced SGTRs are considered in the containment response discussed in Section 3.7.4.

3.7.2 CONTAINMENT ISOLATION ANALYSIS

The internal events IPE containment isolation analysis was used as the basis for the seismic IPEEE containment isolation analysis. Containment isolation is one of the safety functions included in developing the seismic equipment list; therefore, the valves, penetrations, and actuation systems were explicitly included in the seismic walkdowns and in the seismic capacity analysis. Relays associated with the valves and with the containment isolation actuation systems were included in the relay chatter evaluation. Seismic capacity walkdowns were performed to ensure that a seismic event would not cause a unique containment isolation failure. Specific attention was placed on potential spatial interactions and building displacements.

The containment isolation fault tree from the IPE was directly used in the extended accident sequence event trees for the seismic Level 2 analysis. This fault tree contains the penetrations which are connected either to the containment atmosphere (including sump and drain tank lines) or to the RCS. Containment isolation valves power dependencies as well as seismic and random failure modes were directly included in the fault tree quantification. All penetrations have at least one fail-closed isolation valve.

Penetrations not included in the IPE models were also evaluated. Enclosed systems inside containment (e.g. critical loop CCW piping) that were not in the IPE scope were assessed by the seismic IPEEE. All such containment penetrations and their associated valves were evaluated to verify that seismic events with loss of power would not fail the isolation function. Some penetrations included check valves (such as the instrument air supply line) whose function would not be affected by a seismic event. Most of the penetrations had at least one isolation valve that fails closed upon the loss of power.

The remainder of the isolation valves were associated with piping systems which are closed systems inside containment. Those penetrations were specifically examined during the walkdowns to ensure that they would not fail and cause loss of the pressure boundary. For example, the non-critical loop CCW piping which supplies the RCP seal coolers and CEDM cooler was spot-checked as were the coolers themselves. All of the piping and equipment have high seismic capacity (greater than 8g) and would therefore not need to be isolated for a seismic-initiated event. The critical loop CCW piping and heat exchangers inside containment were examined and found to have high capacity (greater than 8g). Seismic failure of the critical loop piping inside containment had been identified as a contributor to early release at another plant. The normal chilled water lines supply water to the normal containment coolers with one fail-closed valve outside containment. The seismic capacity walkdown demonstrated that the valves and coolers have high seismic capacity. Similarly, the main steam lines have fail-closed isolation valves outside containment. Also, the main steam lines have high seismic capacity inside containment. Based on these evaluations and the high capacity of the systems and components, it was not necessary to add these penetrations to the containment isolation fault tree.

The containment isolation actuation signal (CIAS) and containment purge isolation signals (CPIS) are powered by the Class IE 125 V DC buses. A seismic event with loss of AC power would not result in loss of these containment isolation signals. The transmitters and logic cabinets associated with the signals were included in the walkdowns and found to have high seismic capacity. Relays and their cabinets were also included in the evaluation. If 125 V DC power is also lost, then the fail-safe valves would close as designed. Thus, the impacts of the seismic scenarios on the containment isolation actuation function have been evaluated and included in the containment performance analysis.

Three special seismic containment performance issues listed in NUREG-1407 involve penetration cooling requirements, inflatable equipment and personnel hatch seals, and backup air systems for containment isolation valves. None of these issues are significant at SONGS 2/3. Penetration cooling is only needed for lifetime integrity of the penetrations. It is not required for short-term integrity of the penetrations, such as several days or weeks. Inflatable seals are not used at SONGS 2/3. Air-operated containment isolation valves are all fail-closed on loss of air. Backup air is not required.

Based on the above discussion, containment isolation functions, systems and equipment have been reviewed for SONGS 2/3 with no vulnerabilities identified.

3.7.3 CONTAINMENT SYSTEMS PERFORMANCE

In addition to the containment bypass and containment isolation system issues discussed above, containment performance issues also include:

- the potential for gross structural failure of the containment,
- the potential for failure of major equipment or structures inside containment, and
- the potential for failure of components of containment heat removal/pressure suppression systems such as fan coolers, support systems, and system interaction effects (e.g. relay chatter).

Each of these items has been evaluated for the SONGS 2/3 seismic IPEEE. The containment, internal structures, and major equipment (including reactor vessel, RCPs, steam generators, pressurizer, and associated RCS, main steam, and main feedwater piping) have been reviewed (see section 3.5) and found to have high seismic capacity (greater than 8g). While other power plants have found lower capacities for some of these components and structures, the earthquake ground motion, soil-structure interaction, and structural response at SONGS 2/3 result in relatively low demand on these components even at high ground accelerations. The frequency of seismic-induced failure of these structures/components is less than 1E-7/year. Therefore, there are no unique seismic failure modes for these structures and components that need to be incorporated into the containment performance analysis.

Similarly, most equipment on the seismic equipment list which perform containment heat removal/pressure suppression functions have high capacity. The exception are some of the support systems such as electric power and component cooling water. These systems and associated relays have been included in the seismic sequence and scenario analysis discussed in section 3.6. Thus, the seismic-induced failure and non-seismic unavailability of these components have been explicitly included in the seismic containment performance analysis.

The seismic events analysis reported in Section 3.6 can be modeled by one of several accident classes that are counterparts to ones occurring in the IPE analysis, with the additional constraint that systems lost directly by seismic events cannot generally be recovered. These classes are either a loss of offsite power, a loss of the power conversion system, or a small LOCA. Because of subsequent random failures, the loss of offsite power class may lead to a station blackout event, or a small LOCA with loss of power.

Therefore, the seismic impact on containment systems can be assessed by estimating the frequency of occurrence of seismic-induced initiating events of these accident classes, incorporating the specific seismic- induced losses for these scenarios and then evaluating the results using IPE Level 1 and Level 2 models and data. The first two steps have been reported in Section 3.6. The third step, determining the containment response for these conditions, is presented in Section 3.7.4.

For reference, the bounding initiating events by type are:

- Loss of Power Conversion System (PCS) events caused by seismic events identified as:

PCS1:	All Systems Available - (PCS Part 1)
PCS2:	All Systems Available - (PCS Part 2)

- Loss of Offsite Power (LOOP) events caused by seismic events identified as:

LOPAL:	LOOP, with Loss of All AC Power
LOPXT:	LOOP, All Other Systems Available
LOPFX:	LOOP & Turbine Driven AFW Pump Failure

- Station Blackout (SBO) type events subsequent to the Loss of Offsite Power events identified as:

SBOAL:	LOOP, with Loss of All AC Power
SBOXT:	LOOP, All Other Systems Available
SBOFX:	LOOP & Turbine Driven AFW Pump Failure

- Small LOCA (SLOCA) type events identified as:

SLALL:	Earthquake-Induced SLOCA, LOOP & All Other Systems Available
SLTFW:	Earthquake-Induced SLOCA, LOOP & Turbine-Driven AFW Pump Failure

3.7.4 CONTAINMENT RESPONSE TO SEISMIC-CAUSED EVENTS

The SONGS 2/3 containment response assessment in seismic events has found that:

1. There are no new or unique containment failure modes.

2. Seismic events can be considered as resulting in one of several types of initiating events that have already been modeled in the IPE.

These results allow the containment response to be evaluated in the same way as the corresponding internal initiating events.

Similar to the IPE, these dominant seismic damage scenarios were processed through extended event trees, plant damage states and containment event trees. Since the submittal of the IPE, the Level 2 methodology has been upgraded to more directly include estimates of the effects of phenomenological uncertainties on quantification. This expanded Level 2 methodology retains the fundamental SONGS-specific phenomenological understandings presented in the IPE submittal and in responses to NRC Requests for Additional Information (RAI). It is based on the modified REBECA software package for the Level 2 numerical quantifications. NUCAP+ was also employed to solve intermediate steps in quantification of containment phenomenology.

The following containment response outcomes, in percentages of the seismic-induced core damage frequency reported in Section 3.6, have been determined from the quantitative Level 2 assessment. Also included in Table 3.7-1 for comparison is the Level 2 results of the internal events IPE.

TABLE 3.7-1
CONTAINMENT PRESSURE BOUNDARY STATUS DISTRIBUTION OF SEISMIC IPEEE AND IPE

Containment Pressure Boundary Status	Seismic IPEEE Results	Internal Events IPE Results
Containment Not Failed	53%	83%
Leak Type Containment Failure	32%	8%
Rupture Type Containment Failure	13%	4%
Steam Generator Tube Ruptures (Initiated or Induced)	2%	3%
Containment Bypassed (ISLOCA)	None	2%

The containment "not failed" category (53% of the total seismic IPEEE results as stated above) contains the seismic-induced core damage sequences that do not involve over-pressurization failure of containment. However, if long-term recovery actions are not credited, sequences with the potential of late basemat melt-through from core-concrete interactions are included in this category. Due to the significant portion of seismic IPEEE core damage sequences involving loss of all AC power supply (i.e. station blackout) a larger fraction of sequences in this release category have the potential of late basemat melt-through than for the IPE.

The timing of the occurrence of the containment failure categories for the leak and rupture failure modes (excluding the induced SGTR) is:

Time of Containment Failure:	
At or about the time of vessel failure	2%
Late (more than 12 hours later)	43%

The release of radioactivity, using the release categories defined in Section 4.8 of the IPE report [4-4], is distributed among the categories identified in Table 3.7-2.

TABLE 3.7-2
RELEASE CATEGORY AND PROBABILITY OF SEISMIC IPEEE

Release Category	Release Category Definition	Release Frequency (Per Year)	Percent Of Total Core Damage Frequency
S	Success, no containment failure within 48 hours, < 0.1% volatiles released	9.1E-6	53%
L	Late containment failure, up to 1% volatiles released	7.5E-6	43%
B	Containment bypassed, < 0.1% volatiles released	2.6E-7	1.5%
W	Late containment failure, more than 10% volatiles released	2.4E-8	0.2%
G	Early/isolation failure, containment failure prior to or at vessel failure, up to 10% volatiles released	3.9E-7	2.3%
D	Containment bypassed, up to 10% volatiles released	0	0
T	Containment bypassed, > 10% volatiles released	0	0

Two specific Level 2 accident sequences represent 88% of the core damage frequency due to seismic events. Because of this high proportion, essentially all of the calculated results can be interpreted in terms of these two sequences. The first sequence is in the SBOAL-initiated extended event tree which is an SBO sequence with no AC power recovery and failure of the turbine-driven AFW pump to operate. The second sequence is in the SBOXT-initiated extended event tree which is an SBO sequence with the turbine-driven AFW pump operating until battery failure and failure of AC power recovery and 4KV cross-connection to the other unit. The commonality shared by the two sequences is loss of all on-site AC power and failure to restore offsite power within the Level 1 IPE mission time (24 hours). Long-term recovery actions that can reduce the conservatism in the Level 2 analysis are not credited. The core will melt at high RCS pressure with a subsequent induced hot leg/surge line failure highly likely. The debris will fail the vessel and drop into the cavity volume. The water from the RCS will boil off raising the pressure in the containment, and the debris will attack the basemat concrete.

The most important sequences for the small release category S are the two dominant sequences mentioned above which contribute to 82% of the frequency in this category. The most significant cutset (69% of the total) for release category S is the unrecovered loss or failure of all power sources and the turbine-driven auxiliary feedwater pump. The next most significant cutset (6% of the total) is an unrecovered seismic loss of offsite power event with emergency AC power and auxiliary feedwater available. The operator, however, fails to provide long term CST makeup per procedure and core cooling is lost. In both scenarios, the core melts with the RCS at high pressure with the likely consequence of induced hot leg piping failure. In the latter case, safety injection begins and arrests the core melt in-vessel and prevents containment failure, thereby limiting the source term to a small "S" release. In the former case, core cooling is unavailable and the debris attacks the basemat.

The two dominant SBO class sequences contribute 98% of the Category L (typically late failures with dry cavities) releases. The first and second most significant cutset (57% and 25% of the total, respectively) are the same set of events as the dominant S release scenario except that a late overpressure failure is assessed to occur before basemat failure. The third most significant cutset, at 4% of the total, is an unrecoverable seismic loss of offsite power combined with common cause non-seismic failure of four diesel generators leading to the same core melt and containment events as in the preceding scenario.

The most significant contributing sequence (96%) to Category W, late containment failures with a flooded cavity, is a loss of offsite power sequence with loss of auxiliary feedwater with all other mitigating systems available (LOPXT extended event tree). The most significant cutsets (31% and 21%) for this sequence are an unrecovered seismic loss of offsite power event with emergency AC power and auxiliary feedwater available. However, in this case, the operator fails to provide CST makeup per procedure and core cooling is lost. In both scenarios, the core melts with the RCS at high pressure with the likely consequence of induced hot leg piping failure. However, core melt is not arrested in-vessel and the debris falls into the cavity and is covered by water. Containment heat removal is not available because of accident conditions. For release category W, a late containment rupture is more likely than a late containment leak.

Category G (early) releases are due to phenomenological assessments of events such as hydrogen burns or direct containment heating occurring at about the time of vessel failure for all core damage sequences and sequences associated with containment isolation failure. Seismic-induced loss of containment isolation sequences amount to about 1% of all seismic core damage frequency. The two overall dominant sequences discussed earlier contribute 56% to this class. Loss of containment isolation sequences contribute to 38% of this class. The most significant cutset (33%) is the unrecovered loss or failure of all power sources and the turbine-driven auxiliary feedwater pump leading directly to core melt and vessel failure.

Category B releases are the induced steam generator tube rupture bypass class of accidents. The most dominant overall sequence, which contributes 87% of this category is an SBO sequence with no AC power recovery and loss of turbine-driven AFW pump.

One major insight that can be drawn regarding the Level 2 effects is that the frequency of seismic-induced bypass sequences is low. When compared to the IPE results, since there are no ISLOCAs or SGTRs directly caused by seismic events, the frequency is low both absolutely and as a fraction of the core damage frequency.

Another important insight is that the proportion of " no containment failure" outcomes of seismic events is less than for the internal events and the fire-induced events. The predominance of station blackout type accidents, with no credit for long-term recovery, results in no mitigation of the accident in-vessel or in-containment.

Most of the containment failures that do occur are late failures. The majority of the containment failures that do occur are leaks rather than ruptures. These results are consistent with the IPE. The contribution of basemat failure to the small release category is higher for seismic events than for fire and internal initiating events. This type of "late and small releases" can be reduced if long-term recovery actions are credited for the Level 2 analysis.

3.8 USI A-45, GI-131, AND OTHER SEISMIC SAFETY ISSUES

This section discusses the following NRC safety issues with respect to seismic risk:

1.	USI A-45	Shutdown Decay Heat Removal Requirements
2.	GI-131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants
3.	USI A-40	Seismic Design Criteria
4.	USI A-46	Verification of Seismic Adequacy of Equipment in Operating Plants
5.	USI A-17	System Interactions in Nuclear Power Plants
6.	USI A-40	Seismic Capability of Large Safety-Related Above-Ground Tanks
7.	-----------	Eastern U.S. Seismicity (Charleston Earthquake) Issue
8.	GI-57	Effects of Fire Protection System Actuation on Safety Related Equipment

3.8.1 USI A-45: DECAY HEAT REMOVAL

The USI A-45 issue is concerned with reliability and potential vulnerabilities in decay heat removal systems, both for internal and external events. For SONGS 2/3, the safety-related decay heat removal systems for the A-45 issue include the auxiliary feedwater system, the high and low pressure safety injection systems, and the containment spray system. Support systems may include electric power, cooling water (chilled water, CCW and SWC), air/nitrogen, and room cooling and ventilation.

For the case of a transient or small LOCA, the AFW system removes decay heat through the steam generators either to the main condenser or to the atmosphere through the atmospheric dump valves or secondary side safety valves. Long-term decay heat removal is provided through the closed loop shutdown cooling system (SDC), utilizing the LPSI pumps and SDC heat exchangers. In case of a LOCA, the HPSI, LPSI, and CS systems can provide primary inventory makeup and decay heat removal during recirculation. Containment heat removal is also available from the containment fan coolers for the LOCA events.

Each of these systems were included in the analysis of potential earthquakes for the IPEEE. Based on the relatively low CDF from earthquakes at SONGS 2/3 and the conservatism included in the seismic modeling process, there are no identified vulnerabilities in the decay heat removal systems. Generally, the sequences leading to potential core damage involve the seismic initiating event and multiple failures of redundant equipment, often requiring failure of potential operator recovery actions. While the AFW system is very important for decay heat removal, the system components and tanks have high seismic capacity. All of the AFW equipment screens out of the seismic analysis, with the exceptions of the condensate storage tanks (median capacity greater than 8g) and the turbine governor valve (median capacity of 5.6g). Because of the high seismic capacity of the diverse motor driven AFW pumps, sequences with seismic failure of the turbine governor valve contributed less than 1E-8/yr to the core damage frequency. Sequences with seismic failure of the CSTs contributed about 5E-7/yr to the overall CDF, which is not significant.

While a seismic-induced LOCA would be very rare, it was analyzed in the seismic IPEEE analysis. The HPSI, LPSI, and CS system, and the RWST all have high seismic capacity, and their direct seismic failure contribute insignificantly to CDF.

Systems that support the front-line decay heat removal systems also have generally high seismic capacity. The dominant seismic failures were discussed in previous sections, with the two dominant failures involving the 480 v MCCs and switchgear (median capacity of 4.5g), and the loss of instrumentation and control assumed if the auxiliary building fails (median capacity of 5.4g). These failures contributed about 1.1E-5/yr to the seismic CDF, which is 2/3 of the CDF due to seismic events. The NRC staff has previously used 3E-5/yr as the criterion for acceptably small decay heat removal risk. Since the SONGS 2/3 seismic risk is relatively low, it is not judged to be a decay heat removal vulnerability.

In summary, a plant-specific systematic evaluation has been performed for SONGS 2/3 to identify any potential vulnerabilities in the decay heat removal systems. No vulnerabilities were identified for seismic initiating events.

3.8.2 GI-131: POTENTIAL SEISMIC INTERACTION INVOLVING THE MOVABLE IN-CORE FLUX MAPPING SYSTEM USED IN WESTINGHOUSE PLANTS

SONGS 2/3 is a Combustion Engineering design plant, and is not subject to this potential seismic interaction. Therefore, this issue is considered closed for SONGS 2/3.

3.8.3 USI A-40: SEISMIC DESIGN CRITERIA

USI A-40 investigates selected areas of the seismic design process. The NRC staff identified alternative approaches to certain design procedures and modifications to the NRC criteria in the Standard Review Plan to reflect the current state of the art and industry practice. The concern for the seismic capacity of safety-related above-ground tank (at the SSE) is included in USI A-46. USI A-40 is not applicable to SONGS 2/3 since SONGS 2/3 are modern design plants, and the seismic design criteria address the issues identified in USI A-40. Therefore, this issue is considered closed for SONGS 2/3.

3.8.4 USI A-46: VERIFICATION OF SEISMIC ADEQUACY OF EQUIPMENT IN OPERATING PLANTS
USI A-17: SYSTEM INTERACTIONS IN NUCLEAR POWER PLANTS
USI A-40: SEISMIC CAPABILITY OF LARGE SAFETY-RELATED ABOVE-GROUND TANKS

The A-46 issue applies to older plants with a construction permit application docketed before 1972 and does not apply to SONGS 2/3. The scope of A-46 has been expanded by the NRC to include the seismic spatial system interaction of USI A-17 and the concern of USI A-40 for the seismic capability of large safety-related above-ground tanks. Spatial interactions were specifically addressed in the seismic capacity walkdowns and checklists, and the large safety-related yard tanks were demonstrated to have high seismic capacity. Therefore, all of these issues have been adequately addressed by the SONGS 2/3 seismic design criteria and methods and by the seismic capacity walkdowns. These issues are considered closed for SONGS 2/3.

3.8.5 EASTERN U.S. SEISMICITY (CHARLESTON EARTHQUAKE) ISSUE

This issue is not applicable to SONGS 2/3 and is considered closed.

3.8.6 GI-57: EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY RELATED EQUIPMENT, AND NUREG/CR-5088, FIRE RISK SCOPING STUDY

GI-57 and the Fire Risk Scoping Study (FRSS) raised questions regarding seismic-fire interactions, and the potential impacts of spurious actuation of fire protection systems.

Seismic-induced fire/flood interaction issues, including spurious actuation of the fire protection systems, were evaluated in detail as discussed in section 3.3.4. These evaluations included issues such as fires due to potential sources of flammable liquids or hydrogen, and floods due to multiple actuations of fire suppression systems. The overall result is that any potential seismic-induced fires or floods will not affect safety equipment needed for shutdown during or after a seismic event, and the issues are considered closed.

Based on the above discussions, all of these issues are considered closed for SONGS 2/3.

3.9 SUMMARY OF SEISMIC ANALYSIS RESULTS

SCE performed the SONGS 2/3 seismic risk analysis using methods consistent with NUREG/CR-2300 and NUREG/CR-4840 which meet the requirements of Generic Letter 88-20, Supplement 4. This includes development of a plant-specific hazard curve, completion of plant-wide walkdowns of all safety equipment, plant-specific fragility analysis, and sensitivity and uncertainty analyses. The analysis resulted in several plant modifications which are included in the final results. These modifications and their status are provided below:

- improvement in the reliability of cross-connecting Units 2 and 3 to allow a unit's emergency diesel generators to supply power to the other unit in the event the other unit has a station blackout (improved 4kV power availability)

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- strengthening of ammonia tank supports (removes ammonia spill hazard)

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- removal of floor grating surrounding AFW valve actuators (allow valve movement without spatial interaction with surrounding grating)

  STATUS: Implementation by December 30, 1995.

- removal of concrete plug surrounding Unit 2 diesel generator fuel oil transfer piping (2) (improves piping's seismic capacity)

  STATUS: Implementation by December 30, 1995.

- fastening adjacent electrical cabinets/panels together (prevent interactions and relay chatter)

  STATUS: Implementation by March 31, 1996.

- stabilizing light fixtures that interact with electrical cabinets

  STATUS: Implementation by December 30, 1995.

After modifications, the mean seismic core damage frequency for SONGS 2/3 is 1.7E-5/year.

3.10 REFERENCES

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3.11 SEISMIC APPENDICES

3.11.1 SEISMIC EQUIPMENT LIST

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
1	HPSI	RWST 2T005	S21204MT005	8.7	0.36	0.42
2	HPSI	RWST 2T006	S21204MT006	8.7	0.36	0.42
3	HPSI	RWST Lo Level Xmitter (RAS)	2LT03051	S	N/A	N/A
4	HPSI	RWST Lo Level Xmitter (RAS)	2LT03052	S	N/A	N/A
5	HPSI	RWST Lo Level Xmitter (RAS)	2LT03053	S	N/A	N/A
6	HPSI	RWST Lo Level Xmitter (RAS)	2LT03054	S	N/A	N/A
7	HPSI	RWST Iso. Vlv 2HV9300	2HV9300	S	N/A	N/A
8	HPSI	RWST Iso. Vlv 2HV9301	2HV9301	S	N/A	N/A
9	HPSI	Pump 2P017, Train A	S21204MP017	S	N/A	N/A
10	HPSI	Pump 2P018 (Swing)	S21204MP018	S	N/A	N/A
11	HPSI	HPSI 2P018 Transfer Switch - (B)	S21804ED004	S	N/A	N/A
12	HPSI	Pump 2P019	S21204MP019	S	N/A	N/A
13	HPSI	Sump Iso. Vlv 2HV9304 ('B')	2HV9304	S	N/A	N/A
14	HPSI	Sump Iso. Vlv 2HV9305 ('A')	2HV9305	S	N/A	N/A
15	HPSI	Loop 1A Inject Line Isolation Vlv	2HV9323	S	N/A	N/A
16	HPSI	Loop 1A Inject Line Isolation Vlv	2HV9324	S	N/A	N/A
17	HPSI	Loop 1B Inject Line Isolation Vlv	2HV9326	S	N/A	N/A
18	HPSI	Loop 1B Inject Line Isolation Vlv	2HV9327	S	N/A	N/A
19	HPSI	Loop 2A Inject Line Isolation Vlv	2HV9329	S	N/A	N/A
20	HPSI	Loop 2A Inject Line Isolation Vlv	2HV9330	S	N/A	N/A
21	HPSI	Loop 2B Inject Line Isolation Vlv	2HV9332	S	N/A	N/A
22	HPSI	Loop 2B Inject Line Isolation Vlv	2HV9333	S	N/A	N/A
23	HPSI	Loop 2 Hot Leg Inj. Isolation Vlv	2HV9420	S	N/A	N/A
24	HPSI	Loop 1 Hot Leg Inj. Isolation Vlv	2HV9434	S	N/A	N/A
25	HPSI	HPSI Injection Flow Transmitter (1A)	2FT0311-2	S	N/A	N/A
26	HPSI	HPSI Injection Flow Transmitter (1B)	2FT0321-1	S	N/A	N/A
27	HPSI	HPSI Injection Flow Transmitter (2A)	2FT0331-1	S	N/A	N/A
28	HPSI	HPSI Injection Flow Transmitter (2B)	2FT0341-2	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
29	LPSI	Pump 2P015	S21204MP015	>10	0.35	0.3
30	LPSI	Pump 2P016	S21204MP016	>10	0.35	0.3
31	LPSI	Mini-flow Valve for 2P015	2HV8162	S	N/A	N/A
32	LPSI	Mini-flow Valve for 2P016	2HV8163	S	N/A	N/A
33	LPSI	LPSI/SDC Flow Control Vlv	2HV0396	S	N/A	N/A
34	LPSI	LPSI/SDC Flow Control Vlv	2HV8160	S	N/A	N/A
35	LPSI	LPSI/SDC Flow Control Vlv	2HV8161	S	N/A	N/A
36	LPSI	LPSI Header to RCS Loop 1A Isol Vlv	2HV9322	S	N/A	N/A
37	LPSI	LPSI Header to RCS Loop 1B Isol Vlv	2HV9325	S	N/A	N/A
38	LPSI	LPSI Header to RCS Loop 2A Isol Vlv	2HV9328	S	N/A	N/A
39	LPSI	LPSI Header to RCS Loop 2B Isol Vlv	2HV9331	S	N/A	N/A
40	CSS	HPSI/LPSI/CS Mini-flow Iso Vlv	2HV9306	S	N/A	N/A
41	CSS	HPSI/LPSI/CS Mini-flow Iso Vlv	2HV9307	S	N/A	N/A
42	CSS	HPSI/LPSI/CS Mini-flow Iso Vlv	2HV9347	S	N/A	N/A
43	CSS	HPSI/LPSI/CS Mini-flow Iso Vlv	2HV9348	S	N/A	N/A
44	CSS	Containment Emerg Sump Iso Vlv	2HV9302	S	N/A	N/A
45	CSS	Containment Emerg Sump Iso Vlv	2HV9303	S	N/A	N/A
46	CSS	Cont Emerg Sump Level Transmitter-1E	2LT9386	S	N/A	N/A
47	CSS	Cont Emerg Sump Level Transmitter-1E	2LT9389	S	N/A	N/A
48	CSS	Containment Spray Pump 2P012	S21206MP012	>10	0.35	0.3
49	CSS	Containment Spray Pump 2P013	S21206MP013	>10	0.35	0.3
50	CSS	SDC Heat Exchanger 2E003	S21206ME003	10	0.3	0.25
51	CSS	SDC Heat Exchanger 2E004	S21206ME004	10	0.3	0.25
52	CSS	Containment Spray Isol Vlv "A"	2HV9367	S	N/A	N/A
53	CSS	Containment Spray Isol Vlv "B"	2HV9368	S	N/A	N/A
54	CSS	SDC HX 2E004 Outlet Vlv to SDCS (A)	2HV8150	S	N/A	N/A
55	CSS	SDC HX 2E004 Inlet Vlv from SDCS (A)	2HV8152	S	N/A	N/A
56	CSS	SDC HX 2E003 Outlet Vlv to SDCS (B)	2HV8151	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
57	CSS	SDC HX 2E003 Inlet Vlv from SDCS (B)	2HV8153	S	N/A	N/A
58	CSS	SDC HX 2E003 CCW Isol Vlv, Tr. A	2HV6501	S	N/A	N/A
59	CSS	SDC 2E003 CCW Isol Vlv Accumulator	S21203MV085	>10	N/A	N/A
60	CSS	SDC HX Isol Vlv 2HV6501 Sol Operator	2HY6501	S	N/A	N/A
61	CSS	SDC HX 2E004 CCW Isol Vlv, Tr. B	2HV6500	S	N/A	N/A
62	CSS	SDC 2E004 CCW Isol Vlv Accumulator	S21203MV086	>10	N/A	N/A
63	CSS	SDC HX Iso Vlv 2HV6500 Sol Operator	2HY6500	S	N/A	N/A
64	CSS	Cont. Emergency Cooling Unit	S21501ME399	S	N/A	N/A
65	CSS	Cont. Emergency Cooling Unit	S21501ME400	S	N/A	N/A
66	CSS	Cont. Emergency Cooling Unit	S21501ME401	S	N/A	N/A
67	CSS	Cont. Emergency Cooling Unit	S21501ME402	S	N/A	N/A
68	CSS	Containment Normal Cooling Unit	S21501ME393	S	N/A	N/A
69	CSS	Containment Normal Cooling Unit	S21501ME394	S	N/A	N/A
70	CSS	Containment Normal Cooling Unit	S21501ME396	S	N/A	N/A
71	CSS	Containment Normal Cooling Unit	S21501ME397	S	N/A	N/A
72	CSS	Containment Normal Cooling Unit	S21501ME398	S	N/A	N/A
73	CSS	CCW to ECU 2ME399 Sup Iso Vlv (A)	2HV6370	S	N/A	N/A
74	CSS	CCW to ECU 2ME399 Disch Iso Vlv (A)	2HV6371	S	N/A	N/A
75	CSS	CCW to ECU 2ME400 Sup Iso Vlv (B)	2HV6368	S	N/A	N/A
76	CSS	CCW to ECU 2ME400 Disch Iso Vlv (B)	2HV6369	S	N/A	N/A
77	CSS	CCW to ECU 2ME401 Sup Iso Vlv (A)	2HV6366	S	N/A	N/A
78	CSS	CCW to ECU 2ME401 Disch Iso Vlv (A)	2HV6367	S	N/A	N/A
79	CSS	CCW to ECU 2ME402 Sup Iso Vlv (B)	2HV6372	S	N/A	N/A
80	CSS	CCW to ECU 2ME402 Disch Iso Vlv (B)	2HV6373	S	N/A	N/A
81	CSS	Cntmt Spray Hdr Flow Transmitter (A)	2FT0338-1	S	N/A	N/A
82	CSS	Cntmt Spray Hdr Flow Transmitter (B)	2FT0348-2	S	N/A	N/A
83	CSS	Containment Spray Headers	Various	S	N/A	N/A
84	CSS	Auxiliary Relay Cabinet	2L493	11.2	0.28	0.28

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
85	SIT	Safety Injection Tank T008 Vent Valve	2HV9345	S	N/A	N/A
86	SIT	Safety Injection Tank T007 Vent Valve	2HV9355	S	N/A	N/A
87	SIT	Safety Injection Tank T009 Vent Valve	2HV9365	S	N/A	N/A
88	SIT	Safety Injection Tank T010 Vent Valve	2HV9375	S	N/A	N/A
89	SIT	Safety Injection Tank 2T007 (B)	S21204MT007	S	N/A	N/A
90	SIT	Safety Injection Tank 2T008 (A)	S21204MT008	S	N/A	N/A
91	SIT	Safety Injection Tank 2T009 (A)	S21204MT009	S	N/A	N/A
92	SIT	Safety Injection Tank 2T010 (B)	S21204MT010	S	N/A	N/A
93	SIT	SIT 2T010 Outlet Vlv to RC Loop 2B (B)	2HV9370	S	N/A	N/A
94	SIT	SIT 2T009 Outlet Vlv to RC Loop 2A (A)	2HV9360	S	N/A	N/A
95	SIT	SIT 2T008 Outlet Vlv to RC Loop 1A (A)	2HV9340	S	N/A	N/A
96	SIT	SIT 2T007 Outlet Vlv to RC Loop 1B (B)	2HV9350	S	N/A	N/A
97	SIT	SIT 2T008 Loop Drain Isol Vlv	2HV9341	S	N/A	N/A
98	SIT	SIT 2T008 Loop Drain Isol Vlv Sol OP	2HY9341	S	N/A	N/A
99	SIT	SIT 2T007 Loop Drain Isol Vlv	2HV9351	S	N/A	N/A
100	SIT	SIT 2T007 Loop Drain Isol Vlv Sol OP	2HY9351	S	N/A	N/A
101	SIT	SIT 2T009 Loop Drain Isol Vlv	2HV9361	S	N/A	N/A
102	SIT	SIT 2T009 Loop Drain Isol Vlv Sol OP	2HY9361	S	N/A	N/A
103	SIT	SIT 2T010 Loop Drain Isol Vlv	2HV9371	S	N/A	N/A
104	SIT	SIT 2T010 Loop Drain Isol Vlv Sol OP	2HY9371	S	N/A	N/A
105	AFW	Condensate Storage Tank 2T120	S21305MT120	8.44	0.35	0.25
106	AFW	Condensate Storage Tank 2T121	S21305MT121	8.7	0.36	0.42
107	AFW	2T-121 Level Transmitter	2LT32041	S	N/A	N/A
108	AFW	2T-121 Level Transmitter	2LT32042	S	N/A	N/A
109	AFW	Turbine AFW Pump 2P140	S21305MP140	>10	N/A	N/A
110	AFW	TDAFW Control Panel	2L298	S	N/A	N/A
111	AFW	AFW Pump 2P140 Turbine	S21305MK007	>10	N/A	N/A
112	AFW	Temp. Suction Strainer (2P140)	---	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
113	AFW	Motor Driven AFW Pump 2P141	S21305MP141	S	N/A	N/A
114	AFW	Temp. Suction Strainer (2P141)	---	S	N/A	N/A
115	AFW	Motor Driven AFW Pump 2P504	S21305MP504	S	N/A	N/A
116	AFW	Temp. Suction Strainer (2P504)	---	S	N/A	N/A
117	AFW	Flow Control Valve (2P141)	2HV4713	S	N/A	N/A
118	AFW	Flow Control Valve (2P504)	2HV4712	S	N/A	N/A
119	AFW	Flow Control Valve (2P140)	2HV4705	S	N/A	N/A
120	AFW	Flow Control Valve (2P140)	2HV4706	S	N/A	N/A
121	AFW	Bypass Control Valve (2P504)	2HV4762	S	N/A	N/A
122	AFW	AFW 2P504 TO E088 Bypass Sol Vlv	2HY47622	S	N/A	N/A
123	AFW	Bypass Control Valve (2P141)	2HV4763	S	N/A	N/A
124	AFW	AFW 2P141 TO E089 Bypass Sol Vlv	2HY47631	S	N/A	N/A
125	AFW	Cont. Isolation Valve (2E088)	2HV4714	S	N/A	N/A
126	AFW	AFW to SG 2E088 2HV4714 Sol Op	2HY47142	S	N/A	N/A
127	AFW	Cont. Isolation Valve (2E088)	2HV4730	S	N/A	N/A
128	AFW	Cont. Isolation Valve (2E089)	2HV4715	S	N/A	N/A
129	AFW	Cont. Isolation Valve (2E089)	2HV4731	S	N/A	N/A
130	AFW	AFW to SG 2E089 2HV4731 Sol Op	2HY47311	S	N/A	N/A
131	AFW	Flow Transmitter (2E088)	2FIT4720	S	N/A	N/A
132	AFW	Flow Transmitter (2E089)	2FIT4725	S	N/A	N/A
133	AFW	Turbine Pump Stop Valve	2HV4716	S	N/A	N/A
134	AFW	Turbine Pump Governor	2SV4700	5.62	0.36	0.28
135	AFW	S/G 2E089 to AFWPT 2K007 Isol Vlv	2HV8200	S	N/A	N/A
136	AFW	S/G 2E088 to AFWPT 2K007 Isol Vlv	2HV8201	S	N/A	N/A
137	AFW	Aux Relay Pnl for 2HV4705, 2HV4730	2MS4705	10.28	0.28	0.28
138	AFW	Aux Relay Pnl for 2HV4706, 2HV4715	2MS4706	10.28	0.28	0.28
139	AFW	Aux Relay Panel for 2HV4716	2MS4716	12.3	0.26	0.28
140	AFW	TDAFW 2HV4716 Control Panel	2L443	11.2	0.28	0.28

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
141	AC	4kV Bus 2A04	S21804ESA04	13.5	0.3	0.33
142	AC	4kV Bus 2A06	S21804ESA06	13.5	0.3	0.33
143	AC	480V Bus 2B04	S21805ESB04	5.1	0.3	0.28
144	AC	480V Bus 2B06	S21805ESB06	5.1	0.3	0.28
145	AC	480V Bus 2BD	S21805ESBD	4.52	0.37	0.25
146	AC	480V Bus 2BE	S21805ESBE	4.52	0.37	0.25
147	AC	480V Bus 2BH	S21805ESBH	4.52	0.37	0.25
148	AC	480V Bus 2BJ	S21805ESBJ	4.52	0.37	0.25
149	AC	480V Bus BQ (Common)	SA1805ESBQ	4.52	0.37	0.25
150	AC	480V Bus 2BRA	S21805ESBRA	4.52	0.37	0.25
151	AC	480V Bus 2BRB	S21805ESBRB	4.52	0.37	0.25
152	AC	480V Bus BS (Common)	SA1805ESBS	4.52	0.37	0.25
153	AC	480V Bus 2BY	S21805ESBY	4.52	0.37	0.25
154	AC	480V Bus 2BZ	S21805ESBZ	4.52	0.37	0.25
155	AC	4kV- 480V Transformer 2B04X	S21805ESB04X	>10	0.3	0.28
156	AC	4kV- 480V Transformer 2B06X	S21805ESB06X	>10	0.3	0.28
157	AC	NML Res. Aux Transformer 2XR1	S21804ETXR1	0.52	0.3	0.45
158	AC	NML Res. Aux Transformer 2XR2	S21804ETXR2	0.52	0.3	0.45
159	AC	Line Voltage Regulator 2T062	S21807ET062	4.52	0.37	0.25
160	AC	Line Voltage Regulator 2T063	S21807ET063	4.52	0.37	0.25
161	AC	Line Regulator 2Y008	S21807EY008	S	N/A	N/A
162	AC	Line Regulator 2Y009	S21807EY009	S	N/A	N/A
163	AC	120VAC Vital Bus 2Y01	S21807EY01	10.5	0.28	0.28
164	AC	120VAC Vital Bus 2Y02	S21807EY02	10.5	0.28	0.28
165	AC	120VAC Vital Bus 2Y03	S21807EY03	10.5	0.28	0.28
166	AC	120VAC Vital Bus 2Y04	S21807EY04	10.5	0.28	0.28
167	AC	120VAC Bus 2Q062	S21807EQ062	S	N/A	N/A
168	AC	120VAC Bus 2Q063	S21807EQ063	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
169	AC	120VAC Transfer Switch 2YVS1	S21807EQ06301	S	N/A	N/A
170	AC	120VAC Transfer Switch 2YVS2	S21807EQ06201	S	N/A	N/A
171	AC	120VAC Transfer Switch 2YVS3	S21807EQ06302	S	N/A	N/A
172	AC	120VAC Transfer Switch 2YVS4	S21807EQ06202	S	N/A	N/A
173	AC	Class 1E Aux Relay Pnl Load Group A	2L420	S	N/A	N/A
174	AC	Class 1E Aux Relay Pnl Load Group B	2L421	S	N/A	N/A
175	AC	120VAC Inverter 2Y01	S21807EY001	10.5	0.28	0.28
176	AC	120VAC Inverter 2Y02	S21807EY002	10.5	0.28	0.28
177	AC	120VAC Inverter 2Y03	S21807EY003	10.5	0.28	0.28
178	AC	120VAC Inverter 2Y04	S21807EY004	10.5	0.28	0.28
179	AC	Res Aux Tran 2XR1 Disconnect Switch	S21802ED001	0.74	0.2	0.34
180	AC	Res Aux Tran 2XR2 Disconnect Switch	S21802ED002	0.74	0.2	0.34
181	AC	Res Aux Tran 2XR3 Disconnect Switch	S21802ED003	0.74	0.2	0.34
182	AC	Switchyard	None	0.74	0.2	0.34
183	AC	Unit Protective Relay Cabinet	2L070	>10	N/A	N/A
184	AC	120VAC Inverter 6	S21807EY006	11.8	0.28	0.28
185	AC	120VAC Inverter 7	S21807EY007	11.8	0.28	0.28
186	DC	125VDC Bus 2D1	S21806EQD1	S	N/A	N/A
187	DC	125VDC Bus 2D2	S21806EQD2	S	N/A	N/A
188	DC	125VDC Bus 2D3	S21806EQD3	S	N/A	N/A
189	DC	125VDC Bus 2D4	S21806EQD4	S	N/A	N/A
190	DC	Battery 2B007	S21806EB007	10	0.3	0.33
191	DC	Battery 2B008	S21806EB008	10	0.3	0.33
192	DC	Battery 2B009	S21806EB009	10	0.3	0.33
193	DC	Battery 2B010	S21806EB010	10	0.3	0.33
194	DC	Battery Charger 2B001	S21806EB001	10	0.3	0.28
195	DC	Battery Charger 2B002	S21806EB002	10	0.3	0.28
196	DC	Battery Charger 2B003	S21806EB003	>10	0.3	0.28

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
197	DC	Battery Charger 2B004	S21806EB004	>10	0.3	0.28
198	DC	125VDC 1E Distribution Panel 2D1P1	S21806EQD1P1	S	N/A	N/A
199	DC	125VDC 1E Distribution Panel 2D2P1	S21806EQD2P1	S	N/A	N/A
200	DC	125VDC 1E Distribution Panel 2D3P1	S21806EQD3P1	S	N/A	N/A
201	DC	125VDC 1E Distribution Panel 2D4P1	S21806EQD4P1	S	N/A	N/A
202	CVCS	VCT to Chg Pp Header Isolation Valve	2LV0227B	S	N/A	N/A
203	CVCS	RWST to Chg Pp Suction Hdr Iso Vlv	2LV0227C	S	N/A	N/A
204	CVCS	Charging PP Inlet Accumulator (2P190)	S21208MW246	S	N/A	N/A
205	CVCS	Charging PP Inlet Accumulator (2P191)	S21208MW247	S	N/A	N/A
206	CVCS	Charging PP Inlet Accumulator (2P192)	S21208MW248	S	N/A	N/A
207	CVCS	Chg Pp Outlet Accumulator (2P190)	S21208MW218	S	N/A	N/A
208	CVCS	Chg Pp Outlet Accumulator (2P191)	S21208MW219	S	N/A	N/A
209	CVCS	Chg Pp Outlet Accumulator (2P192)	S21208MW220	S	N/A	N/A
210	CVCS	Charging Pump 2P190	S21208MP190	S	N/A	N/A
211	CVCS	Charging Pump 2P191	S21208MP191	S	N/A	N/A
212	CVCS	Charging Pump 2P192	S21208MP192	S	N/A	N/A
213	CVCS	Charging To Regen HX Iso Vlv	2HV9200	S	N/A	N/A
214	CVCS	Charging Line to RCS Loop 2A Iso Vlv	2HV9202	S	N/A	N/A
215	CVCS	Charging Line to RCS Loop 1A Iso Vlv	2HV9203	S	N/A	N/A
216	CVCS	RC Loop to Regen HX Iso Vlv	2HV9204	S	N/A	N/A
217	CVCS	Regenerative Heat Exchanger	S21208ME063	S	N/A	N/A
218	CVCS	RCS Letdown to Regen HX Cntrl Vlv	2TV0221	S	N/A	N/A
219	CVCS	Regen HX to Aux Spray Valve	2HV9201	S	N/A	N/A
220	CVCS	Chem & Vol Cntrl Sys Control Panel	2CR058SS03	S	N/A	N/A
221	SWC	SWC Pp 2P112 (A) - U2 Intake	S21413MP112	>10	0.35	0.3
222	SWC	SWC Pp 2P113 (B) - U2 Intake	S21413MP113	>10	0.35	0.3
223	SWC	SWC Pp 2P114 (B) - U3 Intake	S21413MP114	>10	0.35	0.3
224	SWC	SWC Pp 2P307 (A) - U3 Intake	S21413MP307	>10	0.35	0.3

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
225	SWC	SWC and CCW Sys Control Panel	2CR064SS05	S	N/A	N/A
226	SWC	SWC Pp 2P112 Disch Iso Vlv (A)	2HV6200	S	N/A	N/A
227	SWC	2HV6200 Actuator IA Accumulator	S21413MV069	>10	N/A	N/A
228	SWC	2HV6200 Actuator IA Accumulator	S21413MV070	>10	N/A	N/A
229	SWC	SWC 2P112 Disch Vlv 2HV6200 Sol Op	2HY6200	S	N/A	N/A
230	SWC	SWC Pp 2P113 Disch Iso Vlv (B)	2HV6201	S	N/A	N/A
231	SWC	SWC Pp 2P307 Disch Iso Vlv (A)	2HV6202	S	N/A	N/A
232	SWC	2HV6202 Actuator IA Accumulator	S21413MV073	>10	N/A	N/A
233	SWC	2HV6202 Actuator IA Accumulator	S21413MV074	>10	N/A	N/A
234	SWC	SWC Pp 2P114 Disch Iso Vlv (B)	2HV6203	S	N/A	N/A
235	SWC	2HV6203 Actuator IA Accumulator	S21413MV076	>10	N/A	N/A
236	SWC	2HV6203 Actuator IA Accumulator	S21413MV075	>10	N/A	N/A
237	SWC	SWC 2P114 Disch Vlv 2HV6200 Sol Op	2HY6203	S	N/A	N/A
238	SWC	SWC Discharge to Outfall (B)	2HV6495	S	N/A	N/A
239	SWC	SWC Discharge to Outfall (A)	2HV6497	S	N/A	N/A
240	SWC	Saltwater Cyclone Separator, 2P112	S21413MF366	S	N/A	N/A
241	SWC	Saltwater Cyclone Separator, 2P113	S21413MF367	S	N/A	N/A
242	SWC	Saltwater Cyclone Separator, 2P307	S21413MF487	S	N/A	N/A
243	SWC	Saltwater Cyclone Separator, 2P114	S21413MF488	S	N/A	N/A
244	SWC	Vent Float Vlv To Atm, 2P112	S21413MW458	S	N/A	N/A
245	SWC	Vent Float Vlv To Atm, 2P113	S21413MW459	S	N/A	N/A
246	SWC	Vent Float Vlv To Atm, 2P307	S21413MW460	S	N/A	N/A
247	SWC	Vent Float Vlv To Atm, 2P114	S21413MW461	S	N/A	N/A
248	SWC	SW from CCW Overflow to Seawall	2HV6494	S	N/A	N/A
249	SWC	SWC Overflow Block Vlv to Seawall	2HV6496	S	N/A	N/A
250	SWC	Intake Structure Traveling Screens	None	S	N/A	N/A
251	SWC	2HV6201 Actuator IA Accumulator	S21413MV071	>10	N/A	N/A
252	SWC	2HV6201 Actuator IA Accumulator	S21413MV072	>10	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
253	SWC	Discharge Conduit Gate	None	4	0.2	0.2
254	CCW	CCW Pump, 2P024 Train A	S21203MP024	S	N/A	N/A
255	CCW	CCW Pump, 2P025 Train A/B	S21203MP025	S	N/A	N/A
256	CCW	CCW 2P025 Transfer Switch	S21804ED005	S	N/A	N/A
257	CCW	CCW Pump, 2P026 Train B	S21203MP026	S	N/A	N/A
258	CCW	CCW NCL Return Isol Vlv (A)	2HV6218	S	N/A	N/A
259	CCW	NCL Iso Valve 2HV6218 Accumulator	S21203MV081	>10	N/A	N/A
260	CCW	NCL Iso Valve 2HV6218 Accumulator	S21203MV082	>10	N/A	N/A
261	CCW	NCL Iso Vlv 2HV6218 Solenoid Op	2HY6218	S	N/A	N/A
262	CCW	CCW NCL Return Isolation Vlv (B)	2HV6219	S	N/A	N/A
263	CCW	NCL Iso Valve 2HV6219 Accumulator	S21203MV083	>10	N/A	N/A
264	CCW	NCL Iso Valve 2HV6219 Accumulator	S21203MV084	>10	N/A	N/A
265	CCW	NCL Iso Vlv 2HV6219 Solenoid Op	2HY6219	S	N/A	N/A
266	CCW	2P025 Suction Isol. Valve (A)	2HV6222A	S	N/A	N/A
267	CCW	2P025 Suction Isol. Valve (A)	2HV6222B	S	N/A	N/A
268	CCW	2P025 Suction Isol. Valve (B)	2HV6224A	S	N/A	N/A
269	CCW	2P025 Suction Isol. Valve (B)	2HV6224B	S	N/A	N/A
270	CCW	2P025 Discharge Isol. Valve (A)	2HV6226A	S	N/A	N/A
271	CCW	2P025 Discharge Isol. Valve (A)	2HV6226B	S	N/A	N/A
272	CCW	2P025 Discharge Isol. Valve (B)	2HV6228A	S	N/A	N/A
273	CCW	2P025 Discharge Isol. Valve (B)	2HV6228B	S	N/A	N/A
274	CCW	CCW Supply to 2P025 Mtr Clr Iso Vlv (A)	2HV6227	S	N/A	N/A
275	CCW	CCW Supply to 2P025 Mtr Clr Iso Vlv (B)	2HV6229	S	N/A	N/A
276	CCW	Pump 2P024 Mini-flow Isol. Valve	2HCV6537	S	N/A	N/A
277	CCW	Pump 2P025 Mini-flow Isol. Valve	2HCV6538	S	N/A	N/A
278	CCW	Pump 2P026 Mini-flow Isol. Valve	2HCV6539	S	N/A	N/A
279	CCW	2P025 Mini-flow Return Isol. Valve	2HV6220	S	N/A	N/A
280	CCW	2P025 Mini-flow Return Isol. Valve	2HV6221	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
281	CCW	2P025 Mini-flow Return Isol. Valve	2HV6551	S	N/A	N/A
282	CCW	2P025 Mini-flow Return Isol. Valve	2HV6552	S	N/A	N/A
283	CCW	Surge Tank (Train A)	S21203MT003	>10	N/A	N/A
284	CCW	Press Cntl for CCW Surge Tank	2PCV6412-6414	S	N/A	N/A
285	CCW	Surge Tank (Train B)	S21203MT004	>10	N/A	N/A
286	CCW	Press Cntl for CCW Surge Tank	2PCV6418-6420	S	N/A	N/A
287	CCW	Surge Tank Outlet Valve (Train A)	2HV6225	S	N/A	N/A
288	CCW	Surge Tank Outlet Valve (Train B)	2HV6505	S	N/A	N/A
289	CCW	CCW Heat Exchanger (Train A)	S21203ME001	7.12	0.36	0.25
290	CCW	CCW Heat Exchanger (Train B)	S21203ME002	7.12	0.36	0.25
291	CCW	Non-crit. Loop Suction Isol. Valve	2HV6212	S	N/A	N/A
292	CCW	NCL Supply Iso 2HV6212 Accumulator	S21203MV077	>10	N/A	N/A
293	CCW	NCL Supply Iso 2HV6212 Accumulator	S21203MV078	>10	N/A	N/A
294	CCW	CCW NCL Supply Outlet Cont Iso Vlv (A)	2HV6211	S	N/A	N/A
295	CCW	CCW NCL Supply Inlet Cont Iso Vlv (B)	2HV6223	S	N/A	N/A
296	CCW	NCL Iso Vlv Solenoid Operator (B)	2HY6212	S	N/A	N/A
297	CCW	Non-crit. Loop Suction Isol. Valve	2HV6213	S	N/A	N/A
298	CCW	NCL Supply Iso 2HV6213 Accumulator	S21203MV079	>10	N/A	N/A
299	CCW	NCL Supply Iso 2HV6213 Accumulator	S21203MV080	>10	N/A	N/A
300	CCW	CCW NCL Return Inlet Cont Iso Vlv (A)	2HV6236	S	N/A	N/A
301	CCW	CCW NCL Return Outlet Cont Iso Vlv (B)	2HV6216	S	N/A	N/A
302	CCW	NCL Iso Vlv Solenoid Operator	2HY6213	S	N/A	N/A
303	CCW	CCW Surge Tank Backup N2 Cylinders	S22418MV057-070	S	N/A	N/A
304	CCW	Fuel Bldg Post Acc Clean Up Unit	S21504ME370	S	N/A	N/A
305	CCW	Fuel Bldg Post Acc Clean Up Unit	S21504ME371	S	N/A	N/A
306	CCW	Letdown HX CCW Return Iso Vlv (A)	2HV6293A	S	N/A	N/A
307	CCW	Letdown HX CCW Return Iso Vlv (B)	2HV6522A	S	N/A	N/A
308	CCW	Letdown HX CCW Supply Iso Vlv (A)	2HV6293B	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
309	CCW	Letdown HX CCW Supply Iso Vlv (B)	2HV6522B	S	N/A	N/A
310	CCW	Letdown Heat Exchanger	S21208ME062	S	N/A	N/A
311	CCW	Low Pressure Nitrogen Manifold	2/3TCV5560B	S	N/A	N/A
312	CCW	CCW Surge Tank 2T003 Backpress Reg	2PCV6358	S	N/A	N/A
313	CCW	CCW Surge Tank 2T004 Backpress Reg	2PCV6361	S	N/A	N/A
314	CCW	CCW N2 Supply Vlv to 2T003	2PCV5403	S	N/A	N/A
315	CCW	CCW N2 Supply Vlv to 2T004	2PCV5404	S	N/A	N/A
316	CCW	CCW Surge Tank 2T003 MU Supply Vlv	2HV6273	S	N/A	N/A
317	CCW	CCW Srg Tk 2T003 MU Sup Vlv Sol Op	2HY6273	S	N/A	N/A
318	CCW	CCW Surge Tank 2T004 MU Supply Vlv	2HV6278	S	N/A	N/A
319	CCW	CCW Srg Tk 2T004 MU Sup Vlv Sol Op	2HY6278	S	N/A	N/A
320	CCW	CCW Surge Tank 2T003 Level Trans (A)	2LT6498-1	S	N/A	N/A
321	CCW	CCW Surge Tank 2T004 Level Trans (B)	2LT6499-2	S	N/A	N/A
322	CCW	Primary Plant Makeup Storage Tank	SA1415MT055	3.03	0.2	0.2
323	CCW	Primary Plant Makeup Storage Tank	SA1415MT056	3.03	0.2	0.2
324	CCW	CCW Seismic Makeup Pump 2P1018 (A)	S21203MP1018	S	N/A	N/A
325	CCW	CCW Seismic Makeup Pump 2P1019 (B)	S21203MP1019	S	N/A	N/A
326	CCW	PPMU to CCW Tr B M/U Disch Valve	2HV6569	S	N/A	N/A
327	CCW	PPMU to CCW Tr A M/U Disch Valve	2HV6570	S	N/A	N/A
328	CCW	Essential PPMS	2L411	S	N/A	N/A
329	CCW	CCW to ME336 Condenser Lo Flow Trip	2/3FICL6402	S	N/A	N/A
330	CCW	CCW to ME335 Condenser Lo Flow Trip	2/3FICL6408	S	N/A	N/A
331	MSS	MS ADV (A)	2HV8419	>10	N/A	N/A
332	MSS	MS ADV 2HV8419 Sol Operator	2HY8419C	S	N/A	N/A
333	MSS	MS ADV 2HV8419 Press Control	2HY8419A	S	N/A	N/A
334	MSS	MS ADV 2HV8419 Sol Operator	2HY8419B	S	N/A	N/A
335	MSS	S/G 2E089 Steam Flow Element	2FT1011	S	N/A	N/A
336	MSS	Main Steam Safety Valves	2PSV8401-8409	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
337	MSS	MS ADV (B)	2HV8421	>10	N/A	N/A
338	MSS	MS ADV 2HV8421 Sol Operator	2HY8421C	S	N/A	N/A
339	MSS	MS ADV 2HV8421 Press Control	2HY8421A	S	N/A	N/A
340	MSS	MS ADV 2HV8421 Sol Operator	2HY8421B	S	N/A	N/A
341	MSS	S/G 2E088 Steam Flow Element	2FT1021	S	N/A	N/A
342	MSS	MSIV on 2E089	2HV8204	S	N/A	N/A
343	MSS	MSIV 2HV8204 Dump Vlv Sol	2HY8204Y1	S	N/A	N/A
344	MSS	MSIV 2HV8204 Dump Vlv Sol	2HY8204X1	S	N/A	N/A
345	MSS	MSIV 2HV8204 Dump Vlv Sol	2HY8204Y2	S	N/A	N/A
346	MSS	MSIV 2HV8204 Dump Vlv Sol	2HY8204X2	S	N/A	N/A
347	MSS	MSIV 2HV8205 Dump Vlv Sol	2HY8205Y1	S	N/A	N/A
348	MSS	MSIV 2HV8205 Dump Vlv Sol	2HY8205X1	S	N/A	N/A
349	MSS	MSIV 2HV8205 Dump Vlv Sol	2HY8205Y2	S	N/A	N/A
350	MSS	MSIV 2HV8205 Dump Vlv Sol	2HY8205X2	S	N/A	N/A
351	MSS	MSIV on 2E088	2HV8205	S	N/A	N/A
352	MSS	Main Steam Safety Valves	2PSV8410-8418	S	N/A	N/A
353	MSS	Steam Generator 2E088	S21301ME088	S	N/A	N/A
354	MSS	Steam Generator 2E089	S21301ME089	S	N/A	N/A
355	MSS	Steam Generator Water Control Panel	2CR052SS04	>10	N/A	N/A
356	MSS	N2 Supply to ADV 2HV8419	2PCV8463	>10	N/A	N/A
357	MSS	N2 Storage Acc to 2HV8419	S21301MT212	>10	N/A	N/A
358	MSS	N2 Supply to ADV 2HV8421	2PCV8465	>10	N/A	N/A
359	MSS	N2 Storage Acc to 2HV8421	S21301MT213	>10	N/A	N/A
360	MSS	Blowdown Isolation Valve	2HV4053	S	N/A	N/A
361	MSS	Blowdown Isolation Valve	2HV4054	S	N/A	N/A
362	MSS	Blowdown Isol Vlv Sol Op for 2HV4053	2HY4053	S	N/A	N/A
363	MSS	Blowdown Isol Vlv Sol Op for 2HV4054	2HY4054	S	N/A	N/A
364	MSS	Control Panel Area Turbine Control	2CR054SS05	>10	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
365	MSS	MSIV Bypass Vlv	2HV8202	S	N/A	N/A
366	MSS	MSIV Bypass Vlv, Sol Op	2HY8202	S	N/A	N/A
367	MSS	MSIV Bypass Vlv	2HV8203	S	N/A	N/A
368	MSS	MSIV Bypass Vlv, Sol Op	2HY8203	S	N/A	N/A
369	DG	Diesel Generator 2G002	S22420MG002	>10	N/A	N/A
370	DG	Diesel Generator 2G002 Control Panel	2L160	9.2	0.35	0.28
371	DG	Pressure Indication 2G002	2L287	21.8	0.32	0.43
372	DG	2G002 Radiator Fan (20 Cylinder)	S22420ME546	11.6	0.28	0.21
373	DG	2G003 Radiator Fan (16 Cylinder)	S22420ME550	11.6	0.28	0.21
374	DG	DG 2G002 Transformer Panel	2L376	9.2	0.35	0.28
375	DG	DG 2G002 20 Cyl. Eng. Panel	2L286	21.8	0.32	0.43
376	DG	Expansion Tank (2G002)	S22420MT162	S	N/A	N/A
377	DG	Expansion Tank (2G002)	S22420MT190	S	N/A	N/A
378	DG	Fuel Oil Day Tank (2G002)	S22421MT133	S	N/A	N/A
379	DG	Air Receiver (2G002)	S22420MT335	14.87	N/A	N/A
380	DG	Air Receiver (2G002)	S22420MT336	14.87	N/A	N/A
381	DG	Air Receiver (2G002)	S22420MT339	14.87	N/A	N/A
382	DG	Air Receiver (2G002)	S22420MT340	14.87	N/A	N/A
383	DG	Diesel Generator 2G003	S22420MG003	>10	N/A	N/A
384	DG	Diesel Generator 2G003 Control Panel	2L161	9.2	0.35	0.28
385	DG	Pressure Indication 2G003	2L289	21.8	0.32	0.43
386	DG	2G003 Radiator Fan (20 Cylinder)	S22420ME547	11.6	0.28	0.21
387	DG	2G002 Radiator Fan (16 Cylinder)	S22420ME549	11.6	0.28	0.21
388	DG	DG 2G003 Transformer Panel	2L377	9.2	0.35	0.28
389	DG	DG 2G003 20 Cyl. Eng. Panel	2L288	21.8	0.32	0.43
390	DG	Expansion Tank (2G003)	S22420MT161	S	N/A	N/A
391	DG	Expansion Tank (2G003)	S22420MT189	S	N/A	N/A
392	DG	Fuel Oil Day Tank (2G003)	S22421MT134	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
393	DG	Air Receiver (2G003)	S22420MT337	14.87	N/A	N/A
394	DG	Air Receiver (2G003)	S22420MT338	14.87	N/A	N/A
395	DG	Air Receiver (2G003)	S22420MT341	14.87	N/A	N/A
396	DG	Air Receiver (2G003)	S22420MT342	14.87	N/A	N/A
397	DG	Diesel Fuel Transfer Pump for 2T035	S22421MP093	S	N/A	N/A
398	DG	Diesel Fuel Transfer Pump for 2T036	S22421MP094	S	N/A	N/A
399	DG	Diesel Fuel Transfer Pump for 2T036	S22421MP095	S	N/A	N/A
400	DG	Diesel Fuel Transfer Pump for 2T035	S22421MP096	S	N/A	N/A
401	DG	Diesel Fuel Storage Tank	S22421MT035	S	N/A	N/A
402	DG	Fuel Oil Transfer Pp 2G002 Cntrl Panel	2L160A	S	N/A	N/A
403	DG	Diesel Fuel Storage Tank	S22421MT036	S	N/A	N/A
404	DG	Fuel Oil Transfer Pp 2G003 Cntrl Panel	2L161A	S	N/A	N/A
405	DG	2G002 Day Tank 2T133 Level Trans	2LT5970-1	S	N/A	N/A
406	DG	2G003 Day Tank 2T134 Level Trans	2LT5970-2	S	N/A	N/A
407	DG	DG Bldg Emergency Supply Fan	S21503MA274	S	N/A	N/A
408	DG	DG Bldg Emergency Supply Fan	S21503MA275	S	N/A	N/A
409	DG	DG Bldg Emergency Supply Fan	S21503MA276	S	N/A	N/A
410	DG	DG Bldg Emergency Supply Fan	S21503MA277	S	N/A	N/A
411	DG	DG Air Intake Silencer (A)	S22420MF436	S	N/A	N/A
412	DG	DG Air Intake Silencer (B)	S22420MF437	S	N/A	N/A
413	DG	DG Air Intake Silencer (A)	S22420MF440	S	N/A	N/A
414	DG	DG Air Intake Silencer (B)	S22420MF441	S	N/A	N/A
415	DG	DG Exhaust Silencer (A)	S22420MF438	S	N/A	N/A
416	DG	DG Exhaust Silencer (B)	S22420MF439	S	N/A	N/A
417	DG	DG Exhaust Silencer (A)	S22420MF442	S	N/A	N/A
418	DG	DG Exhaust Silencer (B)	S22420MF443	S	N/A	N/A
419	DG	Electrical Mimic Bus Control	2/3CR063	>10	N/A	N/A
420	DG	DG High kW Alarm Panel	2L621	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
421	HVAC	Switchgear HVAC Cooler Unit 2E255 (A)	S21510ME255	10	0.3	0.25
422	HVAC	Switchgear HVAC Cooler Unit 2E257 (B)	S21510ME257	10	0.3	0.25
423	HVAC	Toxic Gas Monitoring Panel	2/3L378	5.17	0.3	0.28
424	HVAC	HVAC Control Panel	2L154	15.73	0.28	0.28
425	HVAC	East ESF SWGR Rm Hi Temp Alarm	2TISH9812	S	N/A	N/A
426	HVAC	West ESF SWGR Rm Hi Temp Alarm	2TISH9819	S	N/A	N/A
427	HVAC	East ESF SWGR Rm Hi Temp Alarm	2TISH9820	S	N/A	N/A
428	HVAC	West ESF SWGR Rm Hi Temp Alarm	2TISH9826	S	N/A	N/A
429	HVAC	Emergency Supply Fan 053 (A)	SA1510MA053	S	N/A	N/A
430	HVAC	Emergency Supply Fan 054 (B)	SA1510MA054	S	N/A	N/A
431	HVAC	Inverter Room 2A Hi Temp Alarm	2TISH9091A	S	N/A	N/A
432	HVAC	Inverter Room 2B Hi Temp Alarm	2TISH9091B	S	N/A	N/A
433	HVAC	Inverter Room 2C Hi Temp Alarm	2TISH9091C	S	N/A	N/A
434	HVAC	Inverter Room 2D Hi Temp Alarm	2TISH9091D	S	N/A	N/A
435	HVAC	Emergency Exhaust Fan 055 (B)	SA1510MA055	S	N/A	N/A
436	HVAC	Emergency Exhaust Fan 056 (A)	SA1510MA056	S	N/A	N/A
437	HVAC	Control Room HVAC Units	Various	S	N/A	N/A
438	HVAC	Emergency Chiller Pump 2P160 (B)	SA1513MP160	S	N/A	N/A
439	HVAC	Emergency Chiller Pump 2P162 (A)	SA1513MP162	S	N/A	N/A
440	HVAC	Chiller ME335 Transfer Switch	SA1804ED006	S	N/A	N/A
441	HVAC	Chiller ME336 Transfer Switch	SA1804ED007	S	N/A	N/A
442	HVAC	Emergency Chiller Unit 2E335 (B)	SA1513ME335	7.3	0.28	0.33
443	HVAC	Emergency Chiller Unit 2E336 (A)	SA1513ME336	7.3	0.28	0.33
444	HVAC	Compression Tank 2T122 (B)	SA1513MT122	>10	N/A	N/A
445	HVAC	Compression Tank 2T123 (A)	SA1513MT123	>10	N/A	N/A
446	HVAC	Charging 2P192 Emergency AC Unit	S21509ME435	S	N/A	N/A
447	HVAC	Charging 2P191 Emergency AC Unit	S21509ME436	S	N/A	N/A
448	HVAC	Charging 2P191 Emergency AC Unit	S21509ME437	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
449	HVAC	Charging 2P190 Emergency AC Unit	S21509ME438	S	N/A	N/A
450	HVAC	HVAC System Control Panel	2CR060SS17	>10	N/A	N/A
451	RCS	Pressurizer Safety Valve	2PSV0200	S	N/A	N/A
452	RCS	Pressurizer Safety Valve	2PSV0201	S	N/A	N/A
453	RCS	Pressurizer	S21201ME087	S	N/A	N/A
454	RCS	Reactor Coolant Pump 1A	S21201MP001	S	N/A	N/A
455	RCS	Reactor Coolant Pump 2B	S21201MP002	S	N/A	N/A
456	RCS	Reactor Coolant Pump 1B	S21201MP003	S	N/A	N/A
457	RCS	Reactor Coolant Pump 2A	S21201MP004	S	N/A	N/A
458	RCS	Reactor Vessel	S21101MV001	S	N/A	N/A
459	RCS	Reactor Vessel Core Barrel	None	S	N/A	N/A
460	RCS	Reactor Core	None	S	N/A	N/A
461	RCS	Reactor Vessel Internals	None	8	0.3	0.25
462	RCS	Reactor Control Elem Drive Mechanism	S21104CEDM	S	N/A	N/A
463	RCS	Quench Tank	S21201MT011	S	N/A	N/A
464	RCS	RCP Oil Drain Collection Tank 2T215	S21201MT215	>10	N/A	N/A
465	RCS	RCP Oil Drain Collection Tank 2T216	S21201MT216	>10	N/A	N/A
466	RCS	Reactor Coolant Drain Tank	S21901MT012	S	N/A	N/A
467	RCS	RX Coolant Cntrl Sys Control Panel	2CR050SS03	>10	N/A	N/A
468	RCS	Reactivity Cntrl Sys Control Panel	2CR051SS03	>10	N/A	N/A
469	RCS	RCP and PPS Control Panel	2CR056SS02	>10	N/A	N/A
470	ESFAS	PZR Ch A WR Pressure Transmitter	2PT0102-1	S	N/A	N/A
471	ESFAS	PZR Ch B WR Pressure Transmitter	2PT0102-2	S	N/A	N/A
472	ESFAS	PZR Ch C WR Pressure Transmitter	2PT0102-3	S	N/A	N/A
473	ESFAS	PZR Ch D WR Pressure Transmitter	2PT0102-4	S	N/A	N/A
474	ESFAS	Cntmt Ch A NR Pressure Transmitter	2PT0351-1	S	N/A	N/A
475	ESFAS	Cntmt Ch B NR Pressure Transmitter	2PT0351-2	S	N/A	N/A
476	ESFAS	Cntmt Ch C NR Pressure Transmitter	2PT0351-3	S	N/A	N/A
477	ESFAS	Cntmt Ch D NR Pressure Transmitter	2PT0351-4	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
478	ESFAS	Cntmt Ch A WR Pressure Transmitter	2PT0352-1	S	N/A	N/A
479	ESFAS	Cntmt Ch B WR Pressure Transmitter	2PT0352-2	S	N/A	N/A
480	ESFAS	Cntmt Ch C WR Pressure Transmitter	2PT0352-3	S	N/A	N/A
481	ESFAS	Cntmt Ch D WR Pressure Transmitter	2PT0352-4	S	N/A	N/A
482	ESFAS	S/G 2E089 Level Transmitter	2LT1113-1	S	N/A	N/A
483	ESFAS	S/G 2E089 Level Transmitter	2LT1113-2	S	N/A	N/A
484	ESFAS	S/G 2E089 Level Transmitter	2LT1113-3	S	N/A	N/A
485	ESFAS	S/G 2E089 Level Transmitter	2LT1113-4	S	N/A	N/A
486	ESFAS	S/G 2E088 Level Transmitter	2LT1123-1	S	N/A	N/A
487	ESFAS	S/G 2E088 Level Transmitter	2LT1123-2	S	N/A	N/A
488	ESFAS	S/G 2E088 Level Transmitter	2LT1123-3	S	N/A	N/A
489	ESFAS	S/G 2E088 Level Transmitter	2LT1123-4	S	N/A	N/A
490	ESFAS	ESF Actuation Sys Aux Cabinet A	2L034	7.72	0.27	0.23
491	ESFAS	ESF Actuation Sys Aux Cabinet B	2L035	7.72	0.27	0.23
492	ESFAS	Eng Safety Features Control Panel	2CR057SS01	>10	N/A	N/A
493	MSIS	SG #1 (2E089) Pressure Transmitter	2PT1013-1	S	N/A	N/A
494	MSIS	SG #1 (2E089) Pressure Transmitter	2PT1013-2	S	N/A	N/A
495	MSIS	SG #1 (2E089) Pressure Transmitter	2PT1013-3	S	N/A	N/A
496	MSIS	SG #1 (2E089) Pressure Transmitter	2PT1013-4	S	N/A	N/A
497	MSIS	SG #2 (2E088) Pressure Transmitter	2PT1023-1	S	N/A	N/A
498	MSIS	SG #2 (2E088) Pressure Transmitter	2PT1023-2	S	N/A	N/A
499	MSIS	SG #2 (2E088) Pressure Transmitter	2PT1023-3	S	N/A	N/A
500	MSIS	SG #2 (2E088) Pressure Transmitter	2PT1023-4	S	N/A	N/A
501	CIS	PZR Steam Sample Isol Vlv	2HV0510	S	N/A	N/A
502	CIS	PZR Steam Sample Isol Vlv	2HV0511	S	N/A	N/A
503	CIS	Letdown Line Cont Isol Vlv	2TV9267	S	N/A	N/A
504	CIS	RCS Letdown to Letdown HX	2HV9205	S	N/A	N/A
505	CIS	RCP Bleed-off to VCT Isol Vlv	2HV9217	S	N/A	N/A
506	CIS	RCP Bleed-off to VCT Isol Vlv	2HV9218	S	N/A	N/A
507	CIS	Loop 1 Hot Leg Sample Isol Vlv	2HV0508	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
508	CIS	Common Hot Leg Sample Isol Vlv	2HV0509	S	N/A	N/A
509	CIS	Loop 2 Hot Leg Sample Isol Vlv	2HV0517	S	N/A	N/A
510	CIS	Cont Air Rad Mon Isol Vlv	2HV7800	S	N/A	N/A
511	CIS	Cont Air Rad Mon Isol Vlv	2HV7801	S	N/A	N/A
512	CIS	Cont Air Rad Emergency Sample Vlv	2HV7816	S	N/A	N/A
513	CIS	Cont Air Rad Mon Isol Vlv	2HV7802	S	N/A	N/A
514	CIS	Cont Air Rad Mon Isol Vlv	2HV7803	S	N/A	N/A
515	CIS	PZR Surge Line Sample Isol Vlv	2HV0512	S	N/A	N/A
516	CIS	PZR Surge Line Sample Isol Vlv	2HV0513	S	N/A	N/A
517	CIS	Cont Sump Vlv to Radwaste Isol Vlv	2HV5803	S	N/A	N/A
518	CIS	Cont Sump Vlv to Radwaste Isol Vlv	2HV5804	S	N/A	N/A
519	CIS	Cont Air Rad Mon Isol Vlv	2HV7805	S	N/A	N/A
520	CIS	Cont Air Rad Mon Isol Vlv	2HV7810	S	N/A	N/A
521	CIS	Cont Air Rad Mon Isol Vlv	2HV7806	S	N/A	N/A
522	CIS	Cont Air Rad Mon Isol Vlv	2HV7811	S	N/A	N/A
523	CIS	Cont Mini-Purge Supply Isol Vlv	2HV9821	S	N/A	N/A
524	CIS	Cont Mini-Purge Supply Isol Vlv	2HV9823	S	N/A	N/A
525	CIS	Cont Mini-Purge Exhaust Isol Vlv	2HV9824	S	N/A	N/A
526	CIS	Cont Mini-Purge Exhaust Isol Vlv	2HV9825	S	N/A	N/A
527	CIS	RCDT PPs Disharge from Cont	2HV7512	S	N/A	N/A
528	CIS	RCDT 2T012 Drain Isol	2HV7513	S	N/A	N/A
529	CIS	RCDT Vent to Waste Gas Header	2HV7258	S	N/A	N/A
530	CIS	RCDT Vent to Waste Gas Header	2HV7259	S	N/A	N/A
531	CIS	Cont Purge Supply Unit A374 Isol Vlv	2HV9949	S	N/A	N/A
532	CIS	Cont Purge Supply Unit A374 Isol Vlv	2HV9948	S	N/A	N/A
533	CIS	Cont Purge Exhaust Unit Isol Vlv	2HV9950	S	N/A	N/A
534	CIS	Cont Purge Exhaust Unit Isol Vlv	2HV9951	S	N/A	N/A
535	CIS	1E Aux Relay Panel, Load Group A	2L344	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
536	CIS	1E Aux Relay Panel, Load Group B	2L345	S	N/A	N/A
537	CIS	Quench Tank Cont. Isolation Valve	2HV0514	S	N/A	N/A
538	CIS	Quench Tank Cont. Isolation Valve	2HV0515	S	N/A	N/A
539	CIS	Quench Tank Cont. Isolation Valve	2HV0516	S	N/A	N/A
540	CIS	SG Secondary Sample Isolation Valve	2HV4057	S	N/A	N/A
541	CIS	SG Secondary Sample Isolation Valve	2HV4058	S	N/A	N/A
542	CIS	Cont H2 Monitor Isolation Valve	2HV0500	S	N/A	N/A
543	CIS	Cont H2 Monitor Isolation Valve	2HV0501	S	N/A	N/A
544	CIS	Cont H2 Monitor Isolation Valve	2HV0502	S	N/A	N/A
545	CIS	Cont H2 Monitor Isolation Valve	2HV0503	S	N/A	N/A
546	MFW	FW & Condensate Control Panel	2CR053SS04	>10	N/A	N/A
547	MISC	Polar Crane	None	S	N/A	N/A
548	MISC	Penetration Seals/Bldg Expansion Joints	---	S	N/A	N/A
549	MISC	Soil Berm around Demin 2T226,268,277	---	S	N/A	N/A
550	MISC	Condenser Expansion Joints	---	2.82	0.3	0.28
551	MISC	Computer Console	2CR055SS21	>10	N/A	N/A
552	MISC	Control Room Recorders	2CR059SS11	>10	N/A	N/A
553	MISC	Operators Desk	2CR065SS18	>10	N/A	N/A
554	MISC	Remote Evacuation Shutdown Panel	2L042	>10	N/A	N/A
555	PPS	Plant Protection System (PPS) Cabinet	2L032	7.72	0.27	0.23
556	PPS	Reactor Trip Switchgear	2L033	8.7	0.33	0.26
557	PPS	Aux Relay Cabinet (NSSS)	2L071	9.3	0.32	0.22
558	PPS	Aux Protective Cab. CPC/CEAC	2L091	7.72	0.27	0.23
559	PPS	Spec 200 Term Cab Procress Chan 1	2L121	S	N/A	N/A
560	PPS	Spec 200 Term Cab Procress Chan 1	2L122	S	N/A	N/A
561	PPS	Spec 200 Term Cab Procress Chan 1	2L123	S	N/A	N/A
562	PPS	Spec 200 Term Cab Procress Chan 1	2L124	S	N/A	N/A
563	PPS	Spec 200 Term Cab Procress Chan 2	2L125	S	N/A	N/A

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
564	PPS	Spec 200 Term Cab Procress Chan 2	2L126	S	N/A	N/A
565	PPS	Spec 200 Term Cab Procress Chan 2	2L127	S	N/A	N/A
566	PPS	Spec 200 Term Cab Procress Chan 2	2L128	S	N/A	N/A
567	PPS	Spec 200 Term Cab Procress Chan 3	2L129	S	N/A	N/A
568	PPS	Spec 200 Term Cab Procress Chan 3	2L130	S	N/A	N/A
569	PPS	Spec 200 Term Cab Procress Chan 4	2L133	S	N/A	N/A
570	PPS	Spec 200 Term Cab Procress Chan 4	2L134	S	N/A	N/A
571	PPS	Spec 200 Term Cab - NSSS	2L137	S	N/A	N/A
572	PPS	Spec 200 Power Supply - Nest Area	2L138	S	N/A	N/A
573	PPS	Spec 200 Power Supply - Nest Area	2L139	S	N/A	N/A
574	PPS	Spec 200 NSSS Proc Ins Pwr Supply	2L140	S	N/A	N/A
575	PPS	Spec 200 Terminal Cabinet	2L141	S	N/A	N/A
576	PPS	Spec 200 Power Supply/Nest Area	2L142	S	N/A	N/A
577	PPS	Spec 200 Power Supply/Nest Area	2L143	S	N/A	N/A
578	PPS	Spec 200 NSSS Proc Ins Pwr Supply	2L144	S	N/A	N/A
579	PPS	Spec 200 Terminal Cabinet	2L145	S	N/A	N/A
580	PPS	Spec 200 Power Supply/Nest Area	2L146	S	N/A	N/A
581	PPS	Spec 200 Power Supply/Nest Area	2L147	S	N/A	N/A
582	PPS	Spec 200 NSSS Proc Ins Pwr Supply	2L148	S	N/A	N/A
583	PPS	Spec 200 Terminal Cabinet	2L149	S	N/A	N/A
584	PPS	Spec 200 Power Supply/Nest Area	2L150	S	N/A	N/A
585	PPS	Spec 200 Power Supply/Nest Area	2L151	S	N/A	N/A
586	PPS	Spec 200 NSSS Proc Ins Pwr Supply	2L152	S	N/A	N/A
587	PPS	NSSS Interface Spec 200	2L188	S	N/A	N/A
588	PPS	QSPDS Cabinets Channel A	2L491A	S	N/A	N/A
589	PPS	QSPDS Cabinets Channel B	2L491B	S	N/A	N/A
590	FPS	Deluge Vlv for CB 70' North Cable Riser	SA2301MU453	5.72	0.25	0.25
591	FPS	Deluge Vlv for CB 50' North Cable Riser	SA2301MU452	5.72	0.25	0.25

TABLE 3.11-1
SEISMIC EQUIPMENT LIST

ITEM	SYSTEM	DESCRIPTION	TAG NUMBER	SA (g)	β_R	β_U
592	FPS	Deluge Vlv for CB 70' South Cable Riser	SA2301MU456	5.72	0.25	0.25
593	FPS	Deluge Vlv for CB 50' South Cable Riser	SA2301MU457	5.72	0.25	0.25
594	TGIS	Bulk Ammonia Tank	SA1319MT257	S	N/A	N/A

3.11.2 SEISMIC FIGURES

FIGURE 3.0-1
SEISMIC ANALYSIS OVERALL METHODOLOGY

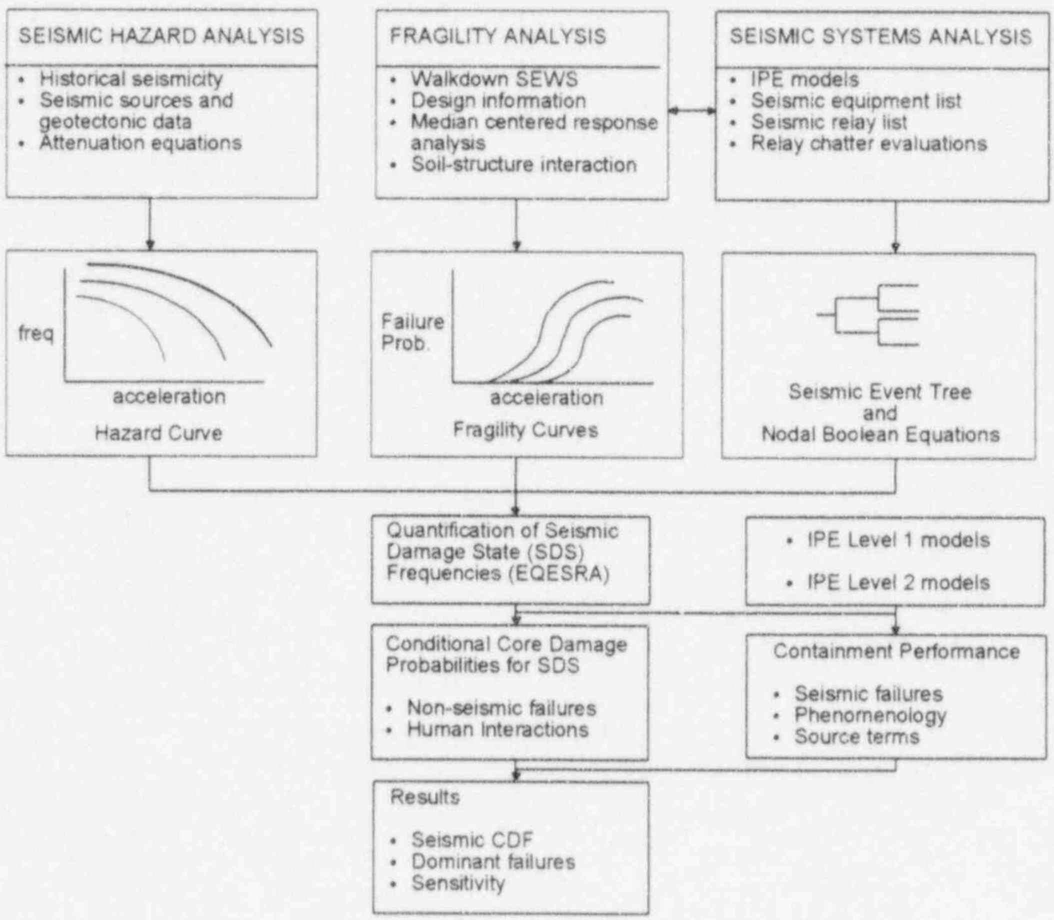


FIGURE 3.1-1
SEISMIC HAZARD COMPUTATIONAL MODEL [3-16]

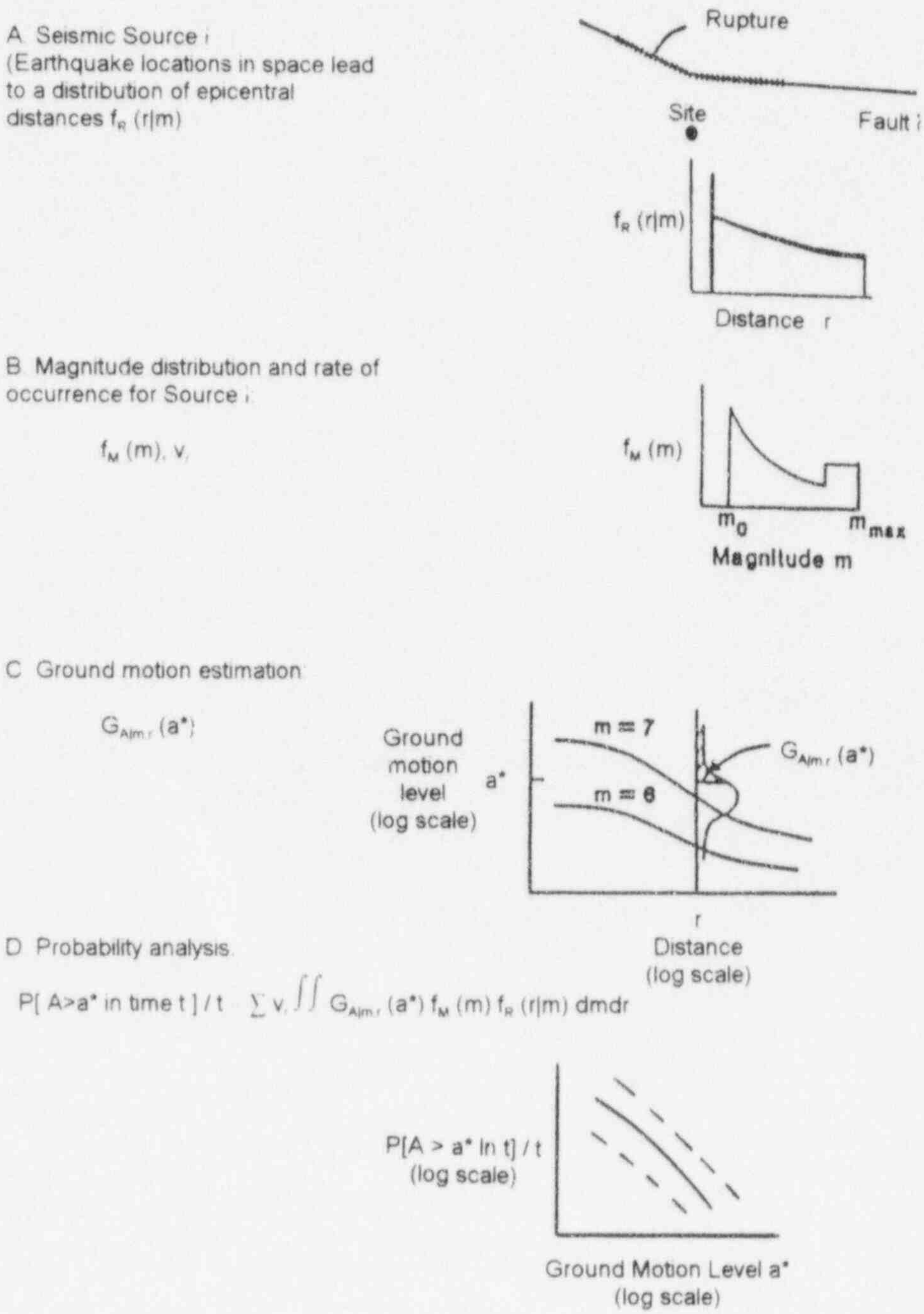
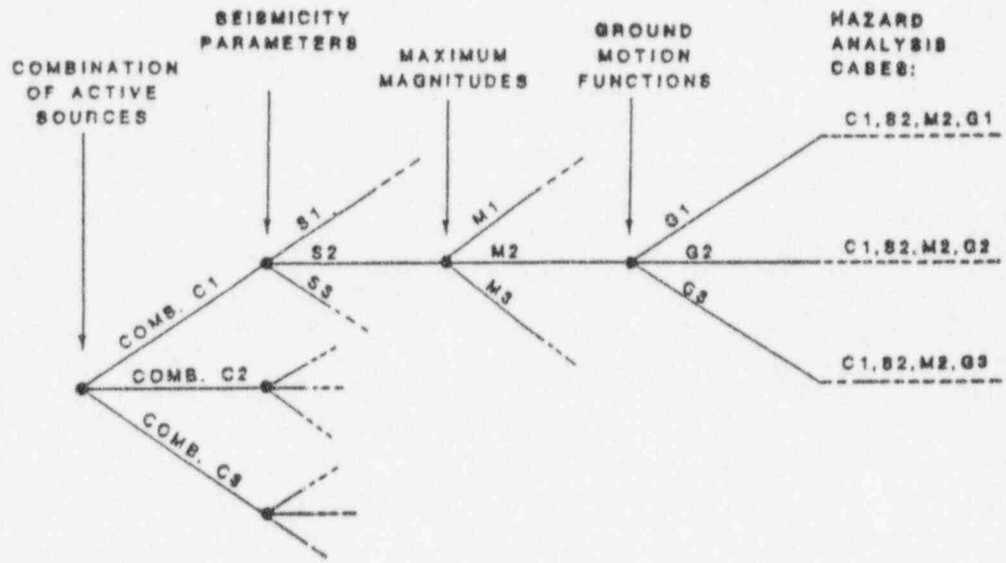


FIGURE 3.1-2
LOGIC TREE REPRESENTATION OF CERTAIN PARAMETERS IN THE EPRI/SOG METHODOLOGY



SOUTHERN CALIFORNIA FAULTS

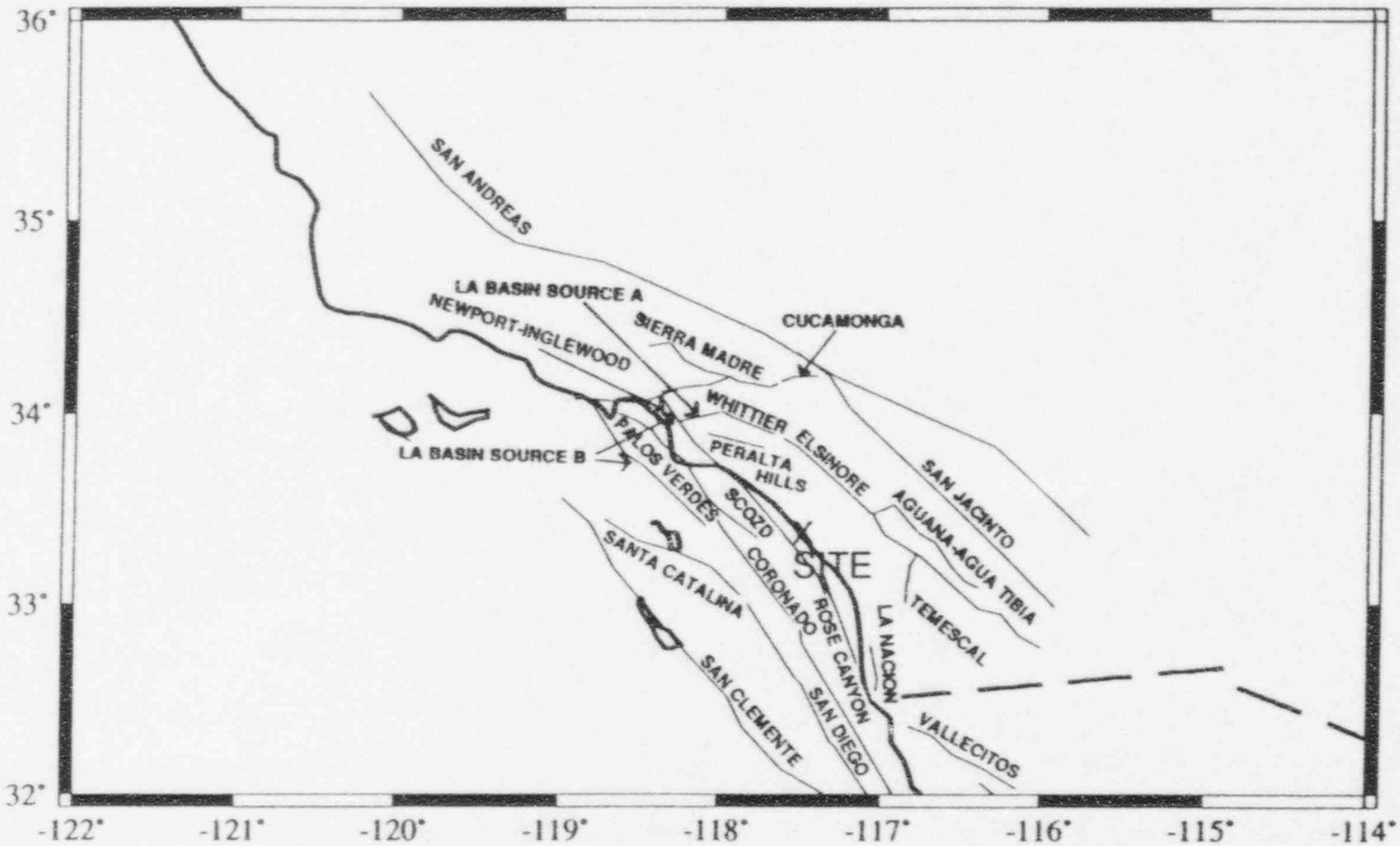


FIGURE 3.1-3
ACTIVE FAULTS IN SOUTHERN CALIFORNIA INCLUDED IN THE STUDY

FIGURE 3.1-4
SOUTHERN CALIFORNIA AREA SOURCES

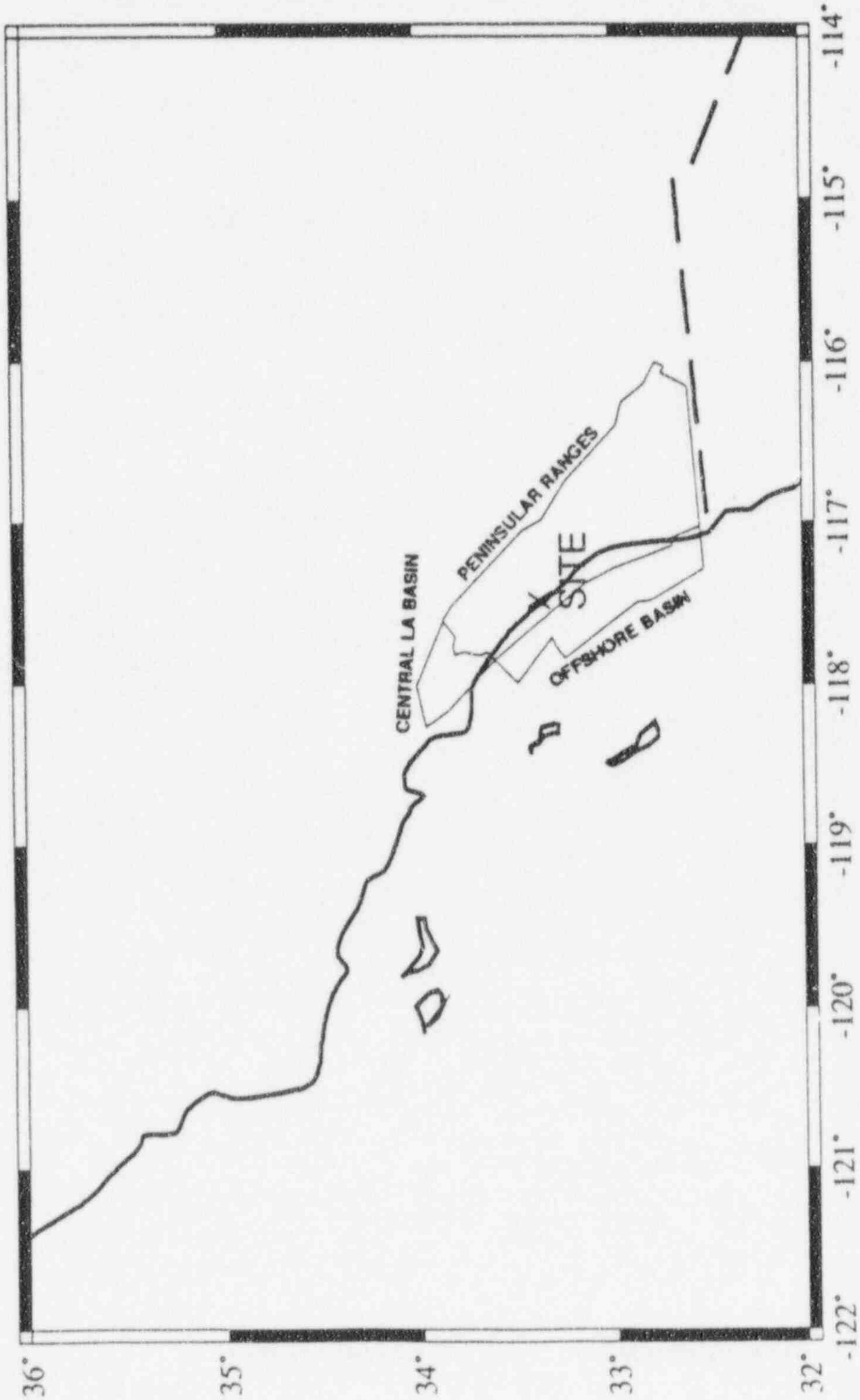


FIGURE 3.1-5
FAULT CORRIDORS USED FOR THE ANALYSIS OF HISTORICAL SEISMICITY IN THIS STUDY

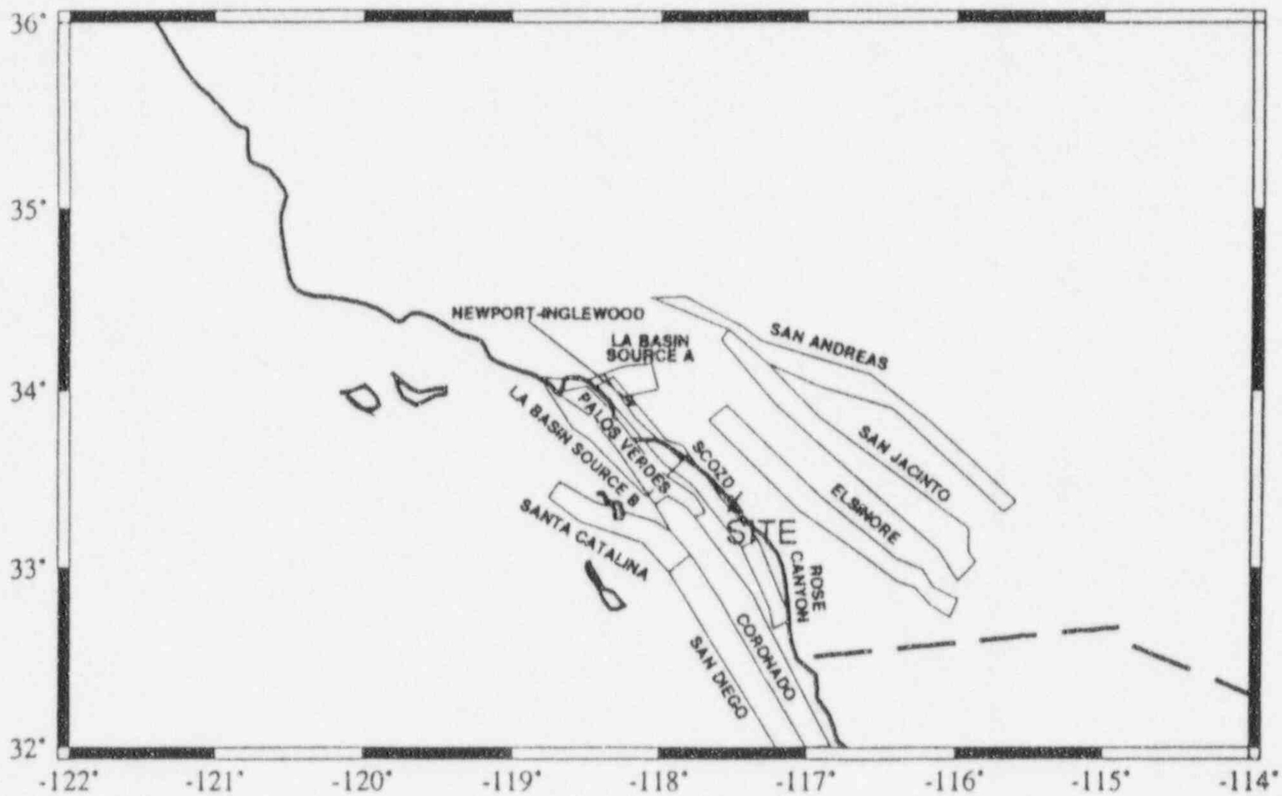


FIGURE 3.1-6
EPICENTERS OF ALL EARTHQUAKES IN SOUTHERN CALIFORNIA CATALOG

FIGURE 3.1-7
EPICENTERS OF DEPENDENT EARTHQUAKES IN SOUTHERN CALIFORNIA

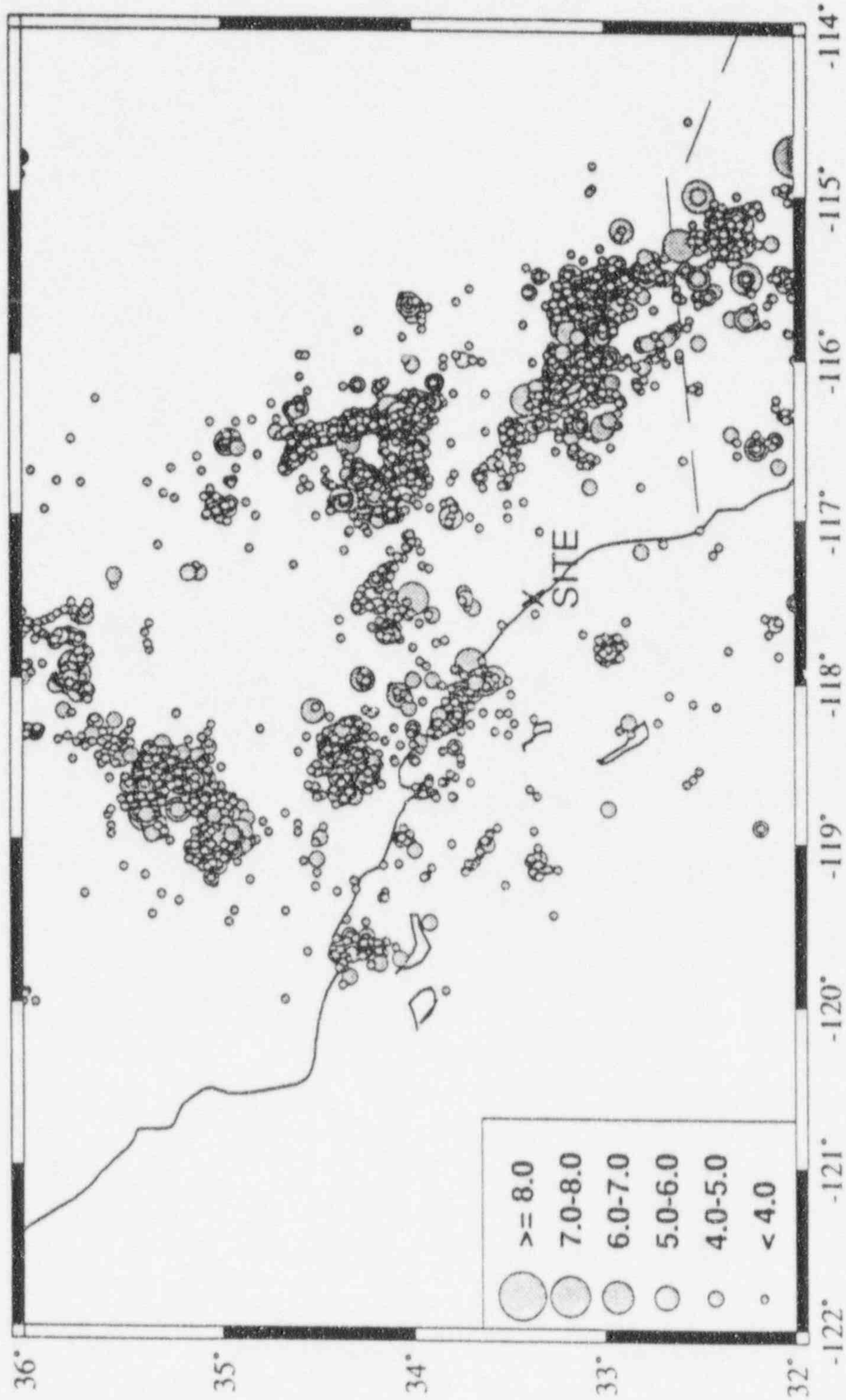


FIGURE 3.1-8
EPICENTERS OF MAINSHOCKS IN SOUTHERN CALIFORNIA

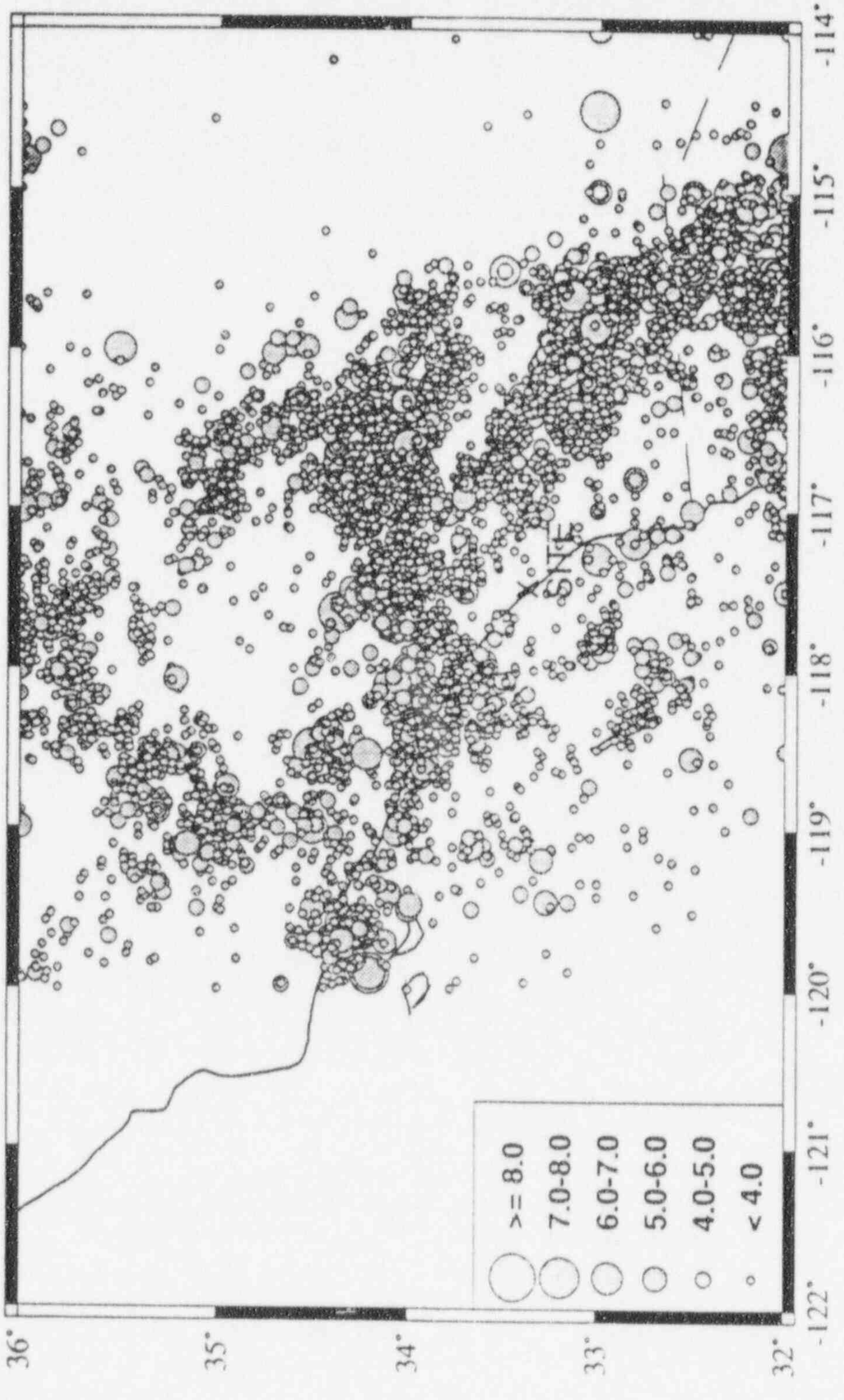


FIGURE 3.1-9
NEWPORT-INGLEWOOD-SCOZD FAULT HISTORICAL SEISMICITY ALONG WITH THE .85, MEDIAN, AND .15 FRACTILES OF THE SEISMICITY ESTIMATED FROM SLIP RATE

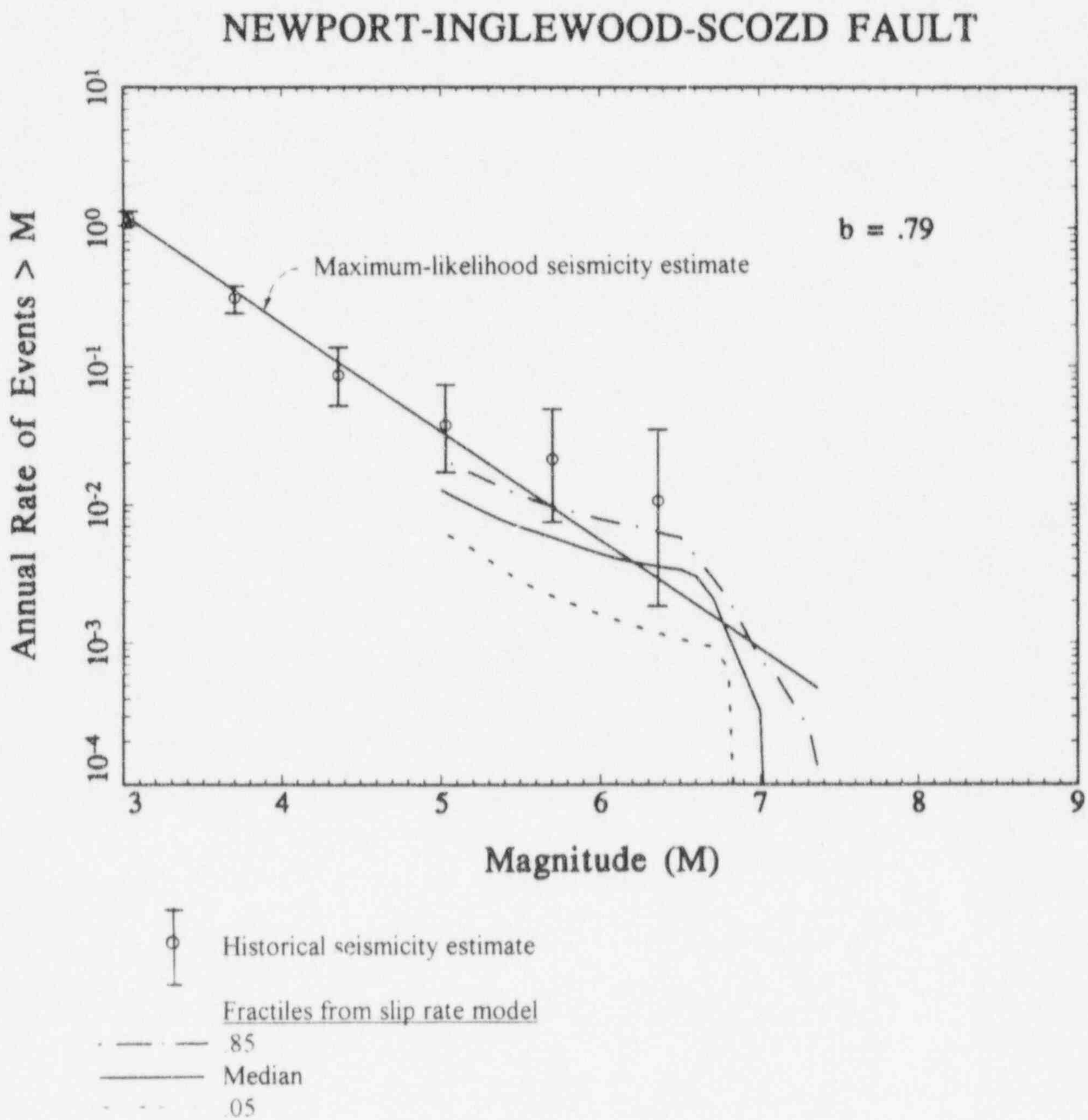


FIGURE 3.1-10
S_a (10 HZ) HAZARD BY FAULT

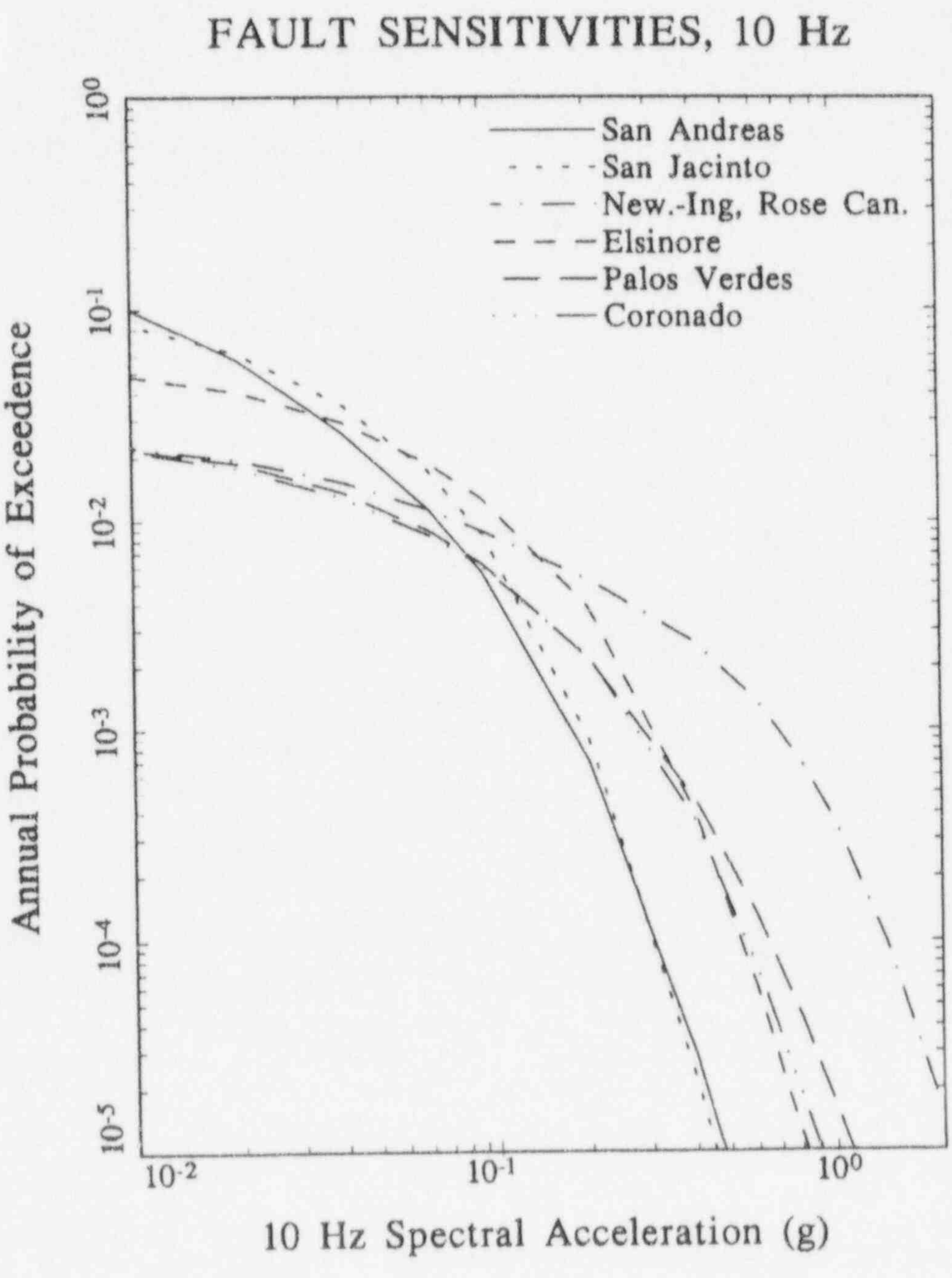


FIGURE 3.1-11
S_a (10 HZ) HAZARD BY FAULT

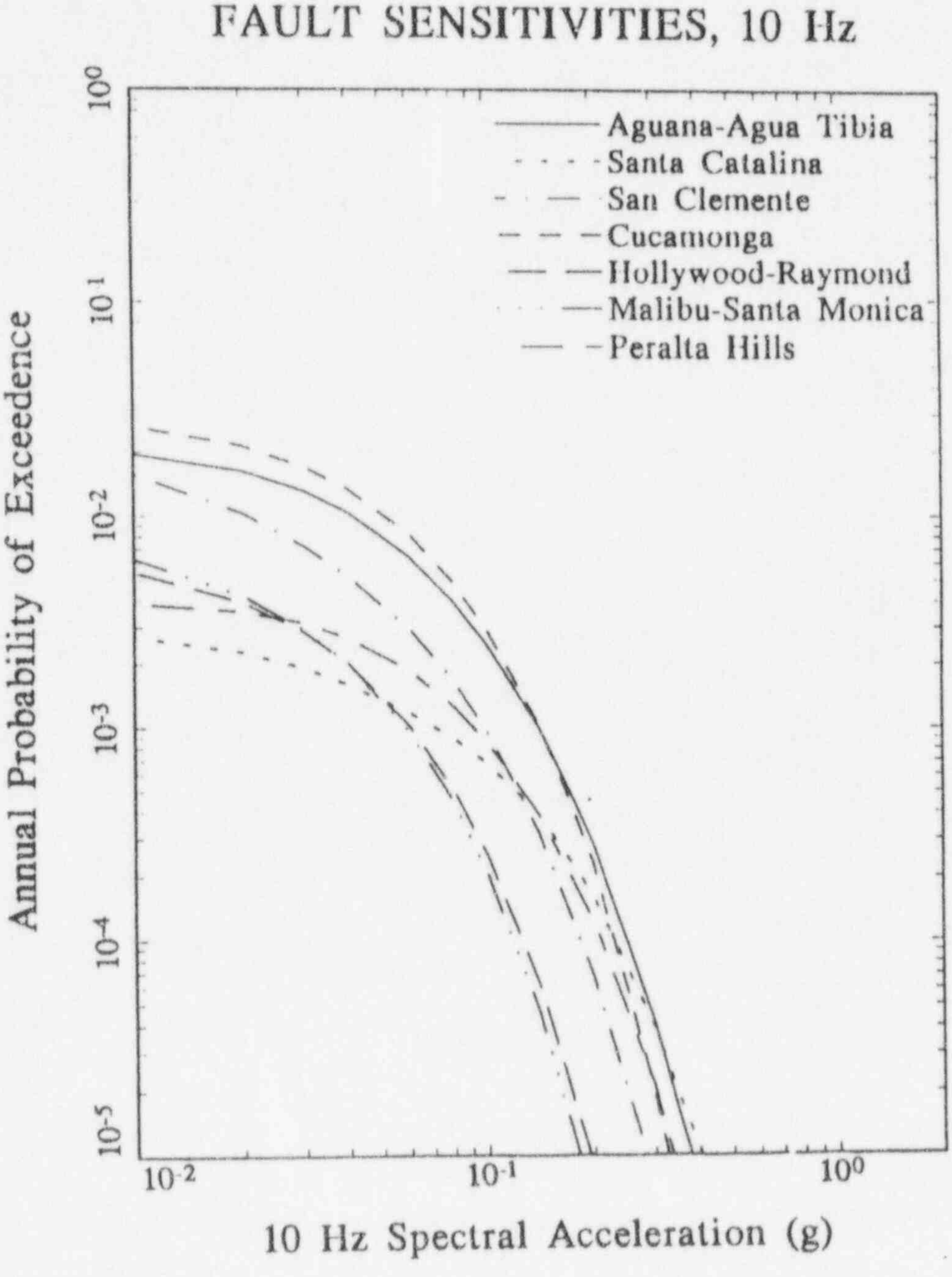


FIGURE 3.1-12
S_a (10 HZ) HAZARD BY FAULT

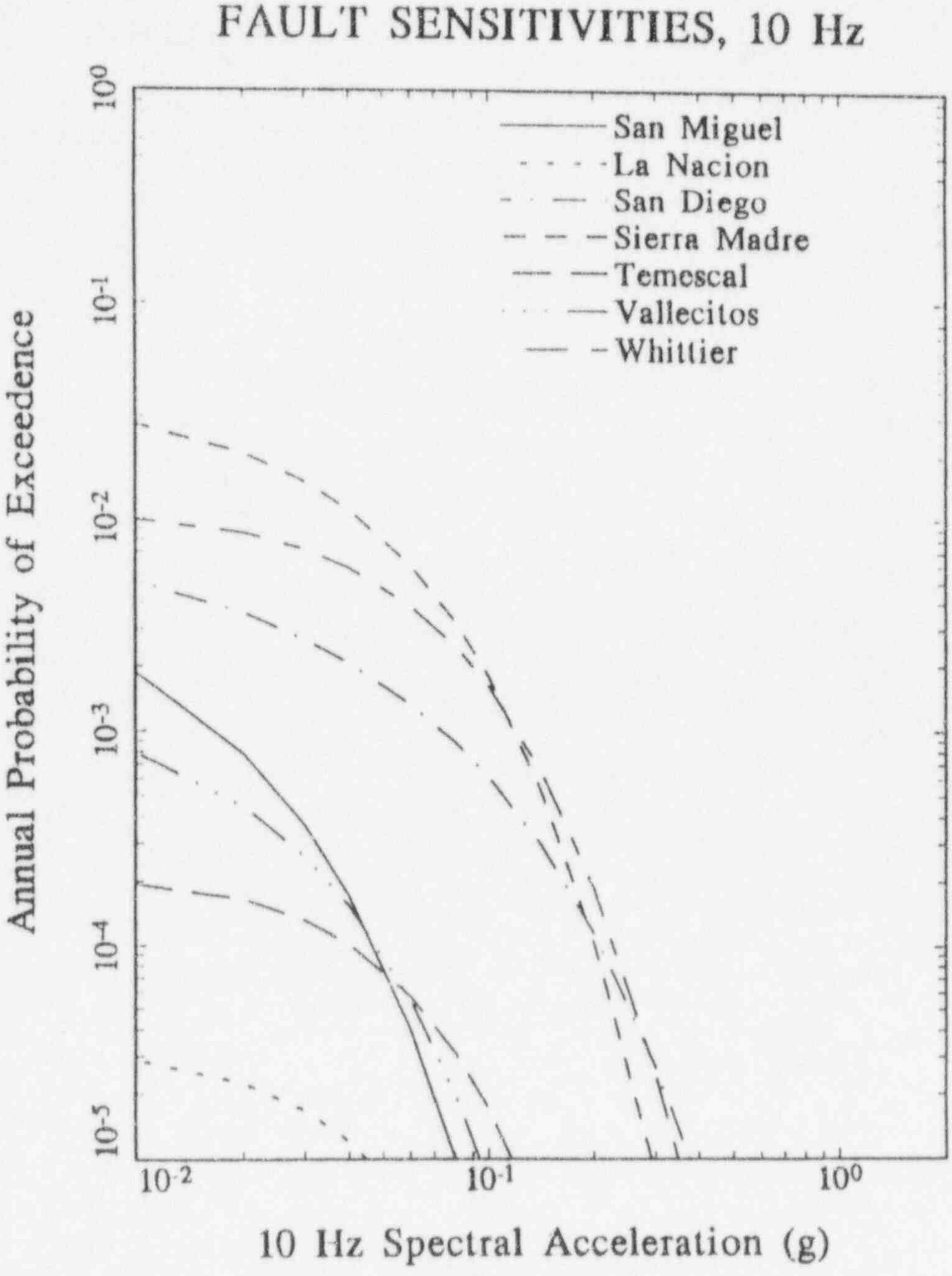


FIGURE 3.1-13
S_a (10 HZ) HAZARD BY BLIND THRUST

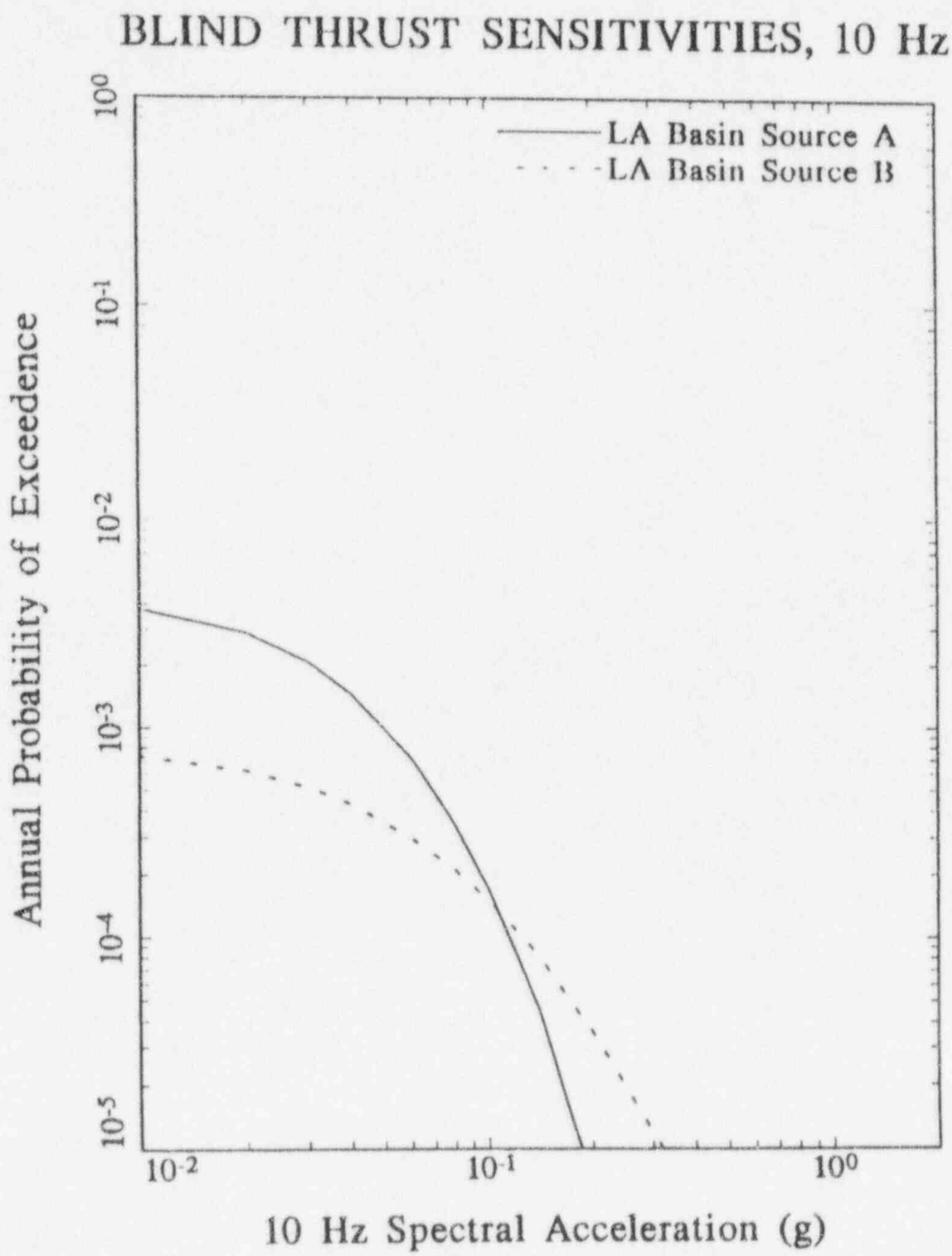


FIGURE 3.1-14
S_a(10 HZ) HAZARD BY AREA SOURCE

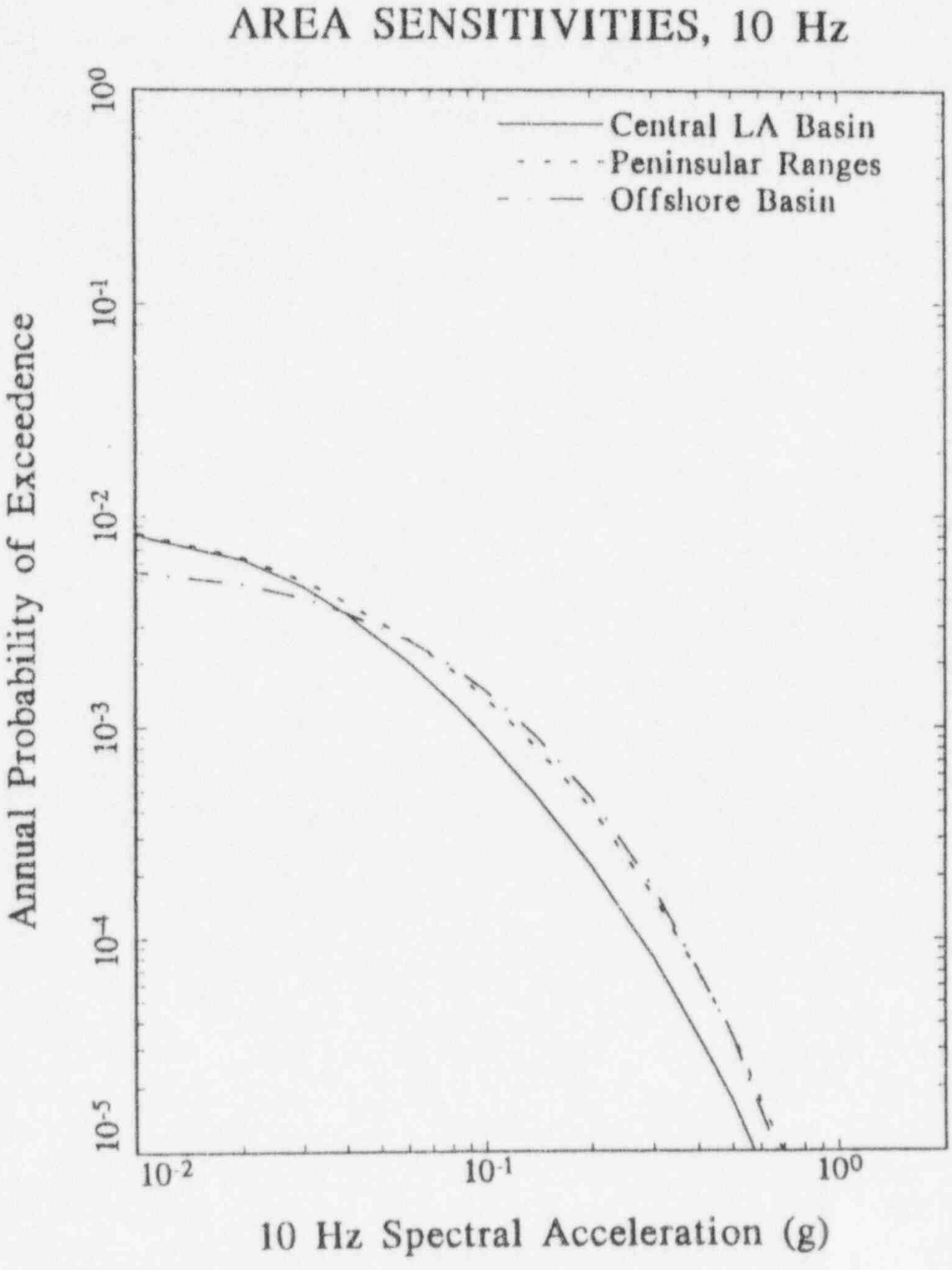


FIGURE 3.1-15
S_a (10 HZ) HAZARD, SENSITIVITY TO ATTENUATION EQUATION

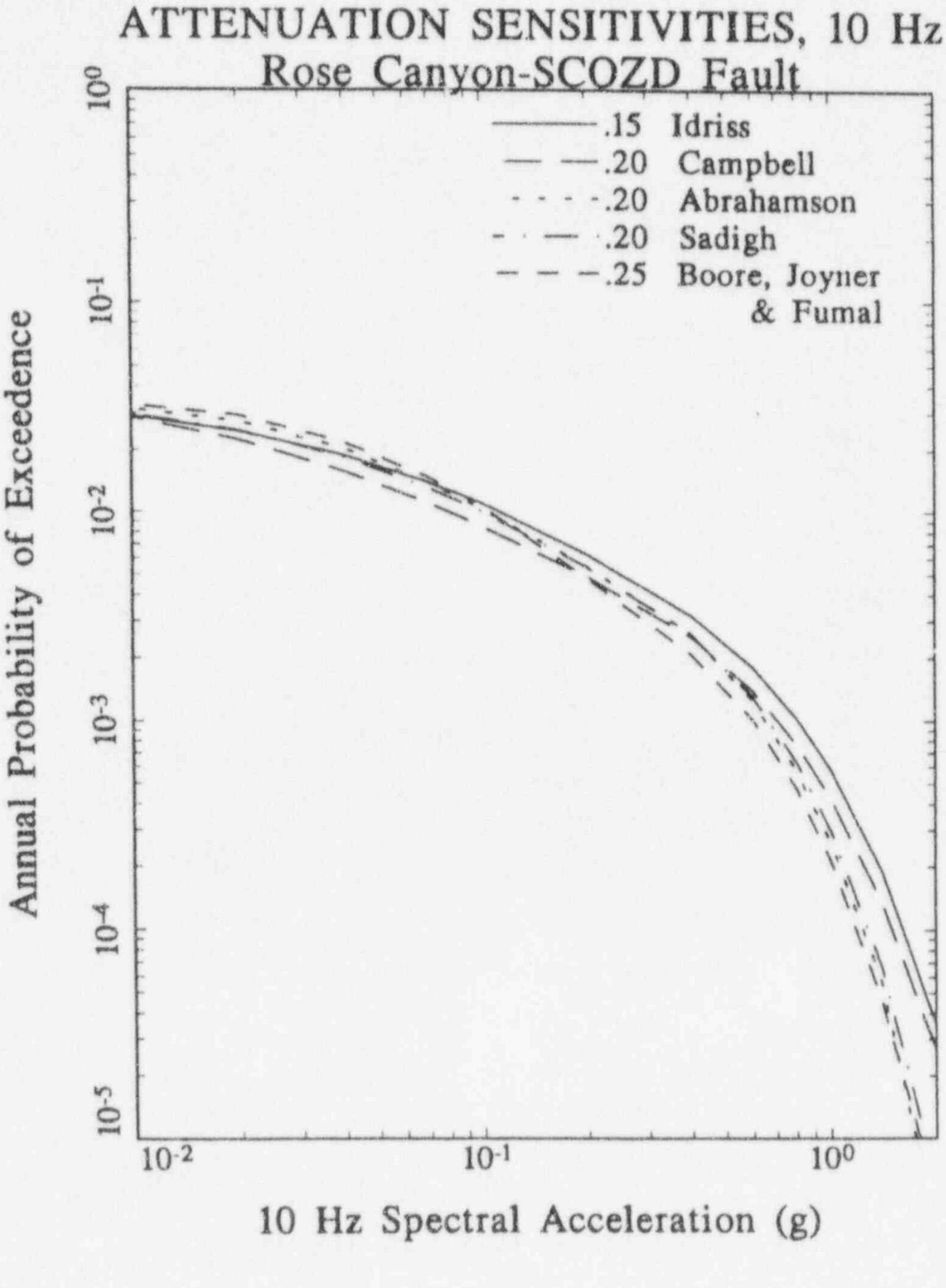


FIGURE 3.1-16
S_a (10 HZ) HAZARD, SENSITIVITY TO SLIP RATE

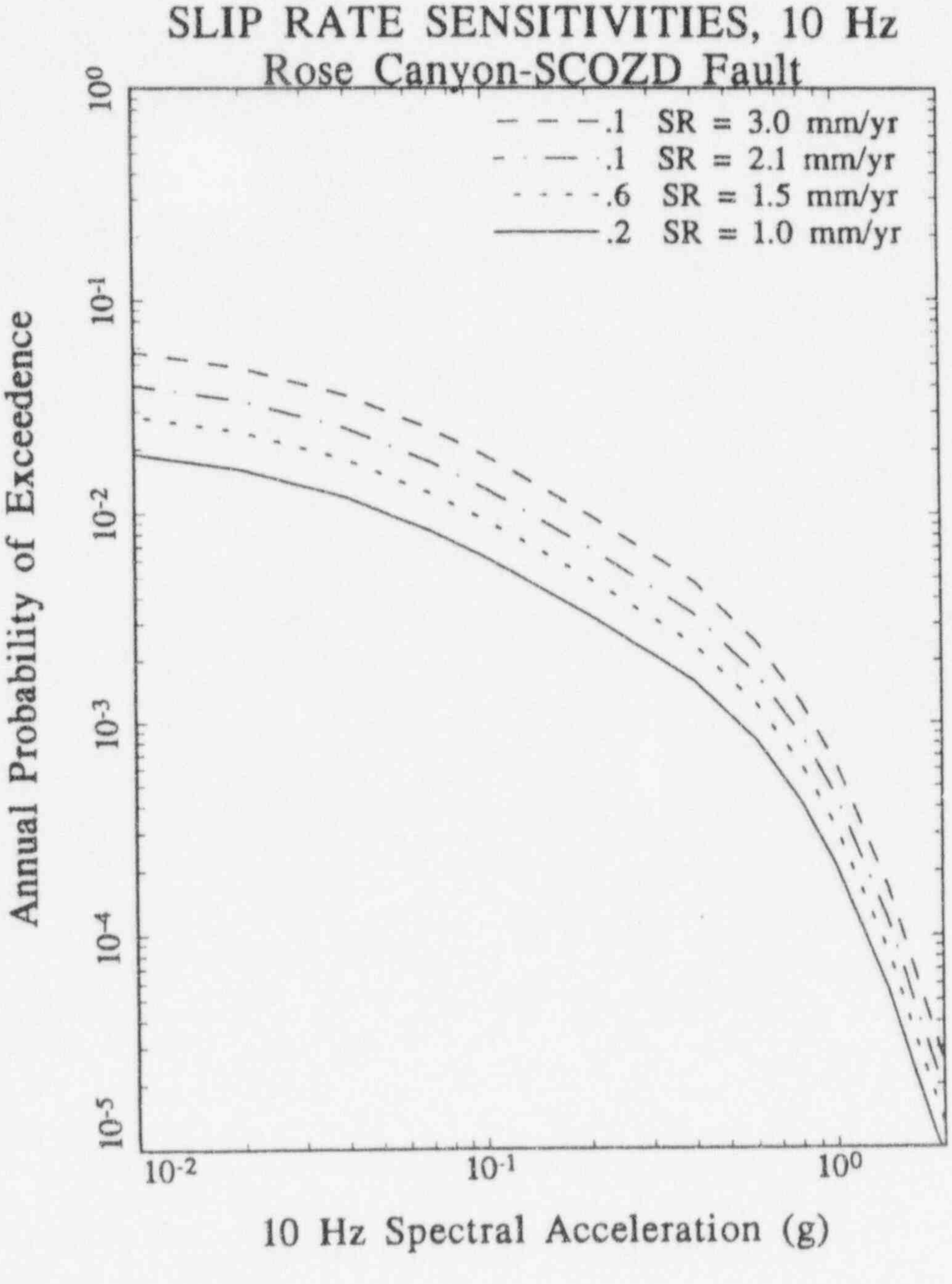


FIGURE 3.1-17
S_a (10 HZ) HAZARD, SENSITIVITY TO b-VALUES

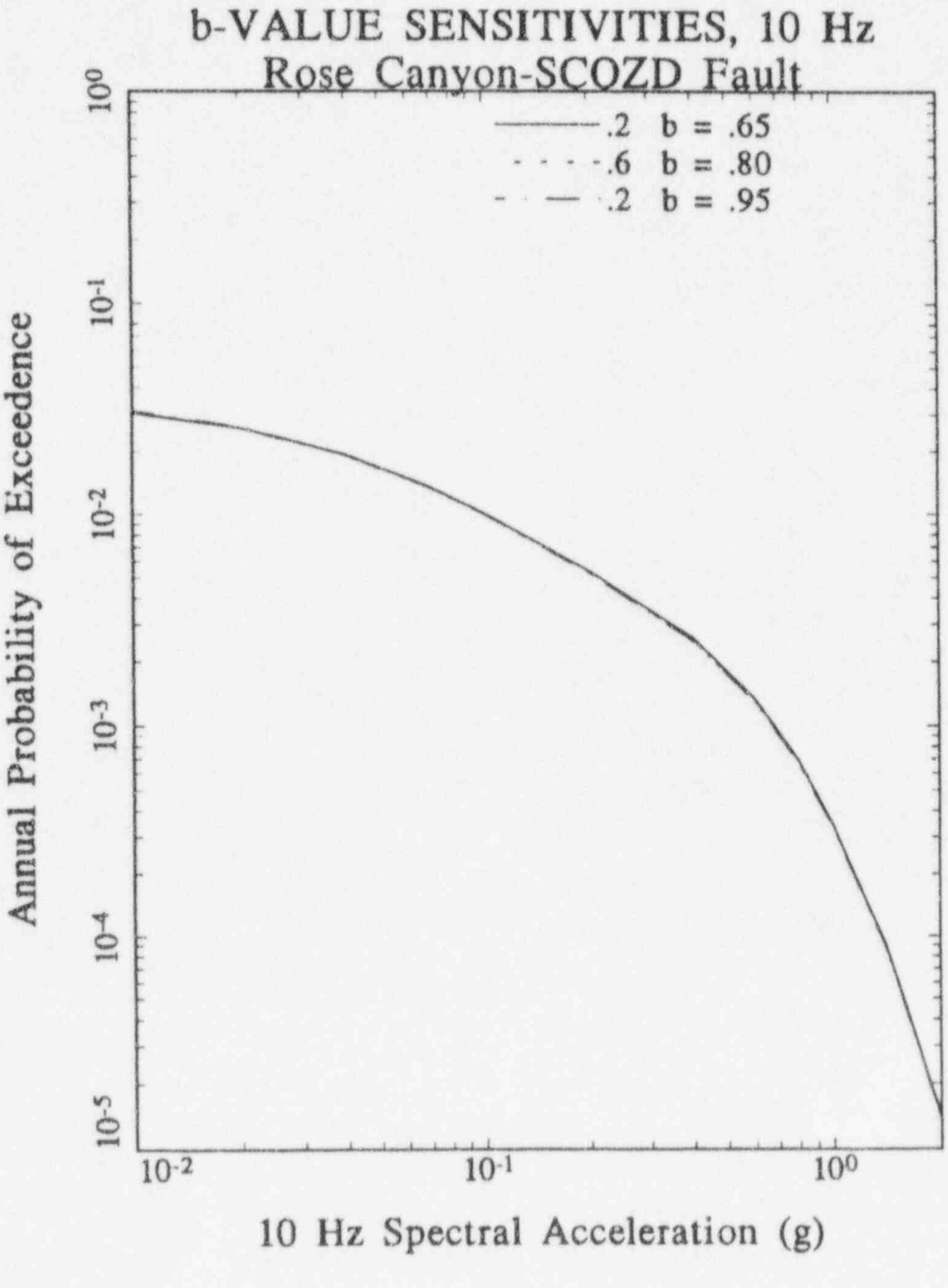


FIGURE 3.1-18
S_a (10 HZ) HAZARD, SENSITIVITY TO DEPTH

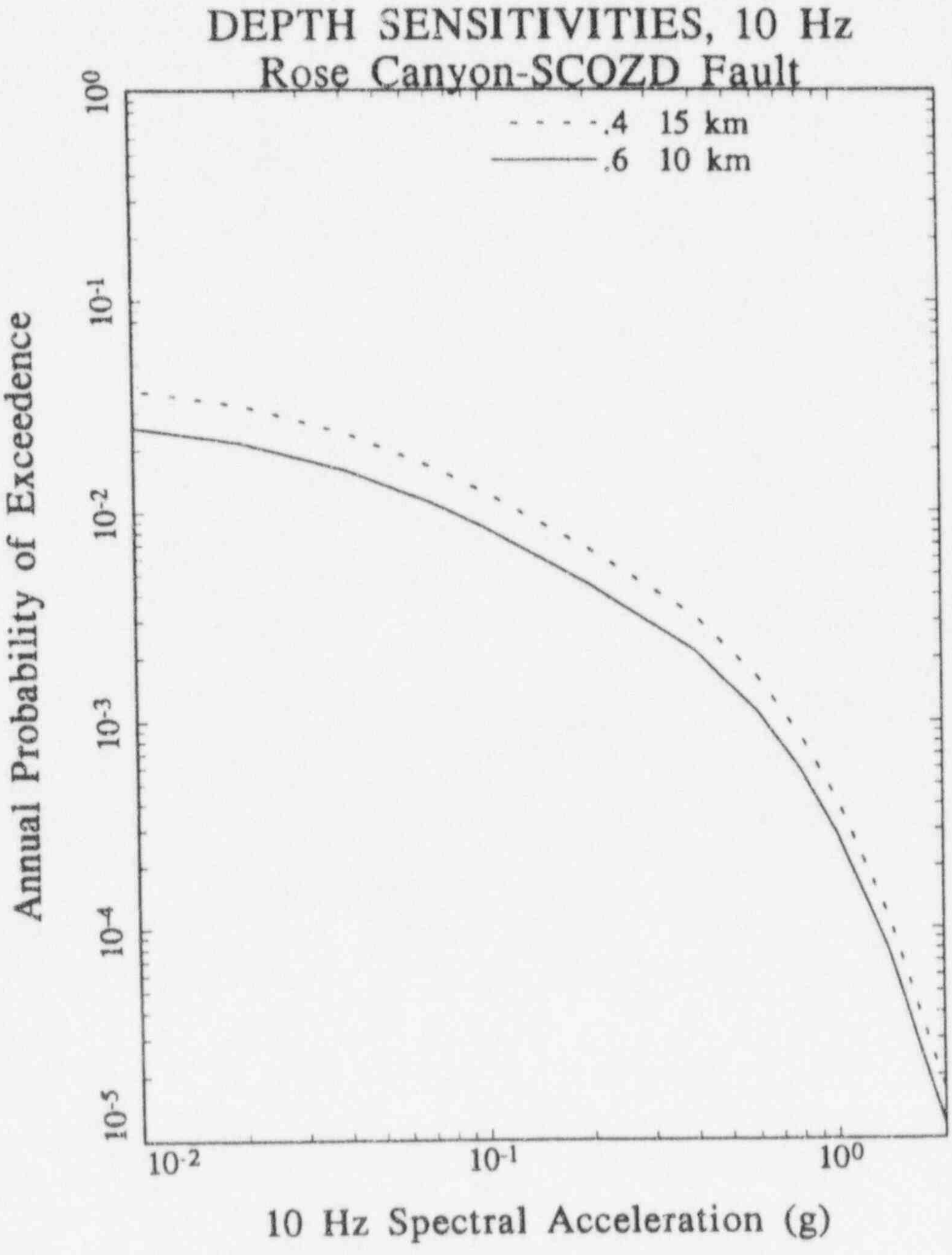


FIGURE 3.1-19
S_a (10 HZ) HAZARD, SENSITIVITY TO MAXIMUM MAGNITUDE

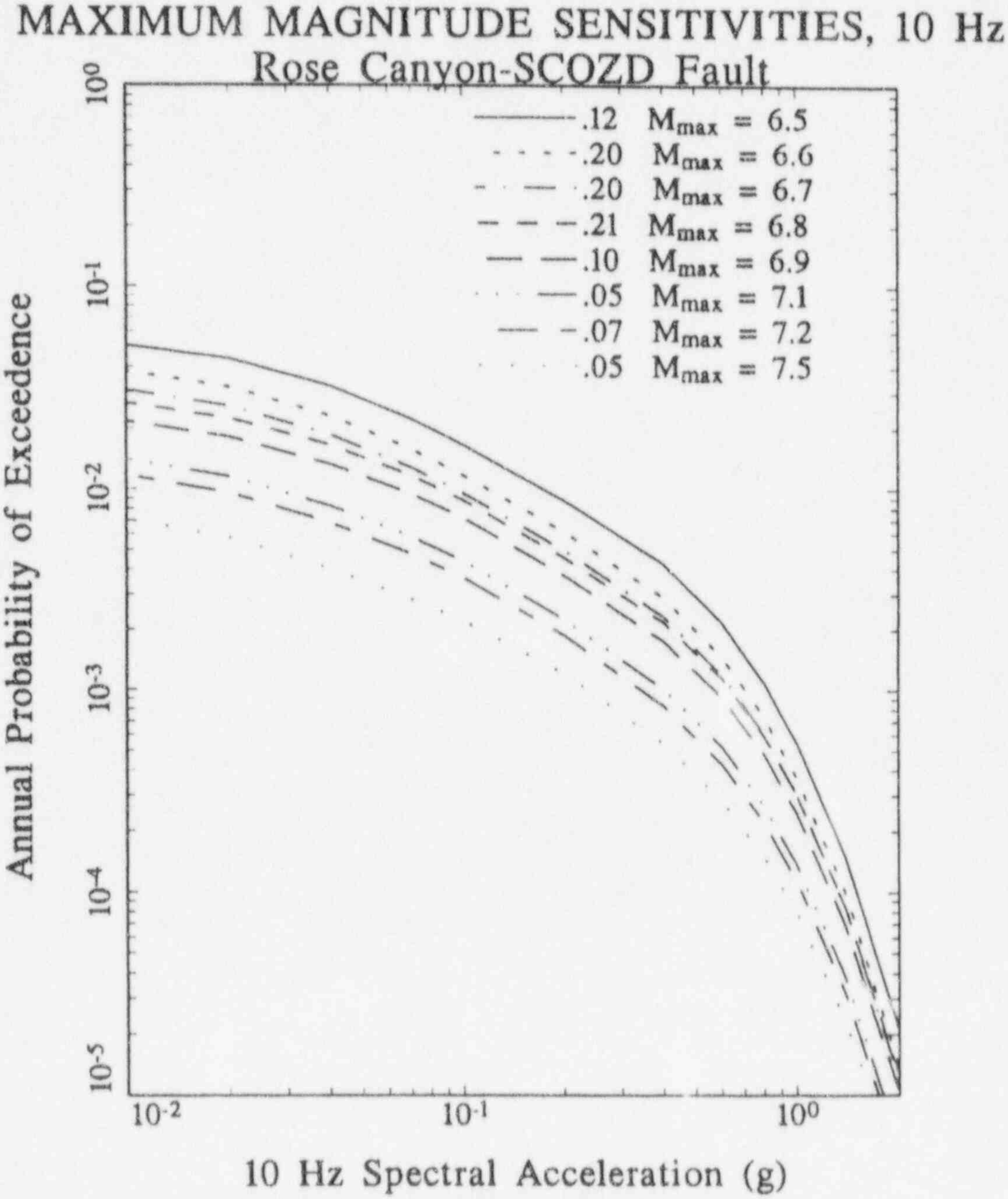


FIGURE 3.1-20
MEAN AND FRACTILE SEISMIC HAZARD CURVES FOR AVERAGE SPECTRAL ACCELERATION, 1 TO 10 HZ

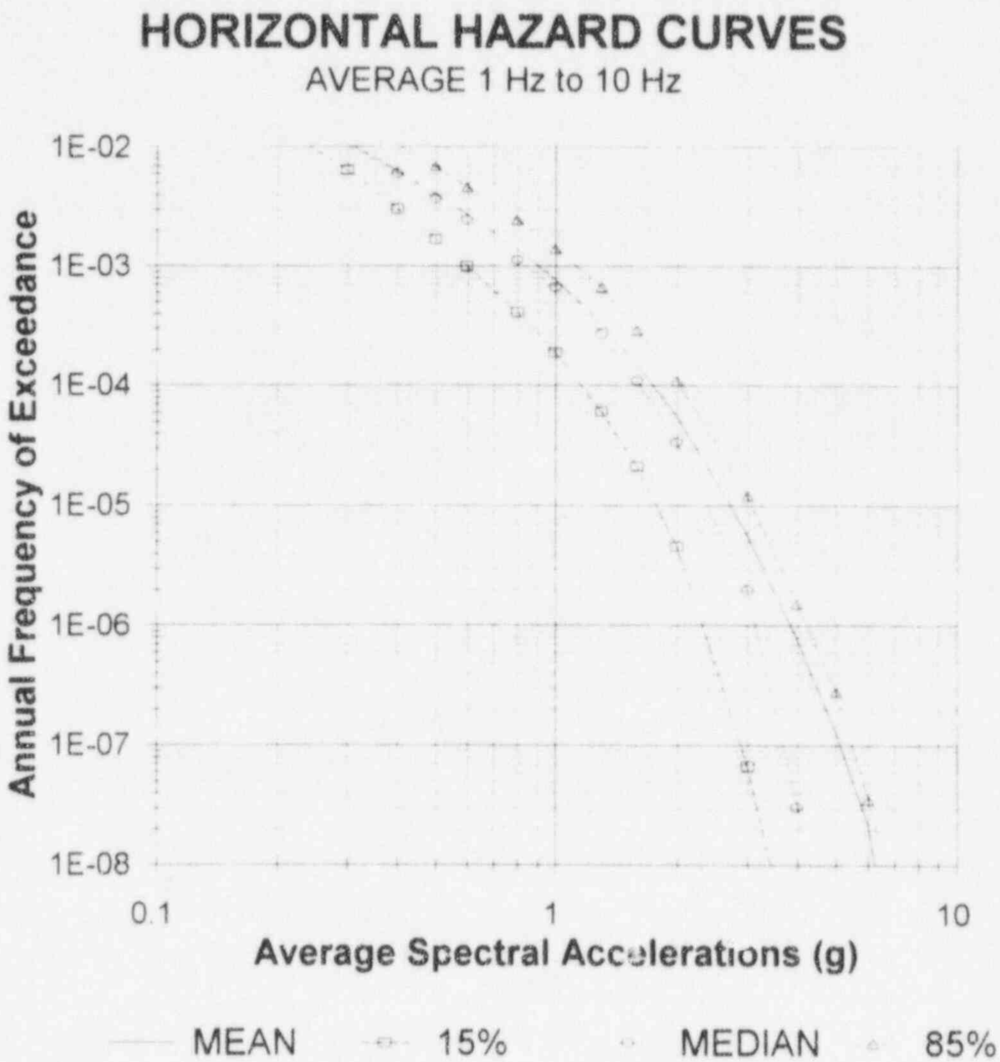


FIGURE 3.1-21
FRACTILES AND MEAN OF UNIFORM HAZARD SPECTRA
$1.4X10^{-4}$ (SSE) ANNUAL PROBABILITY OF EXCEEDENCE

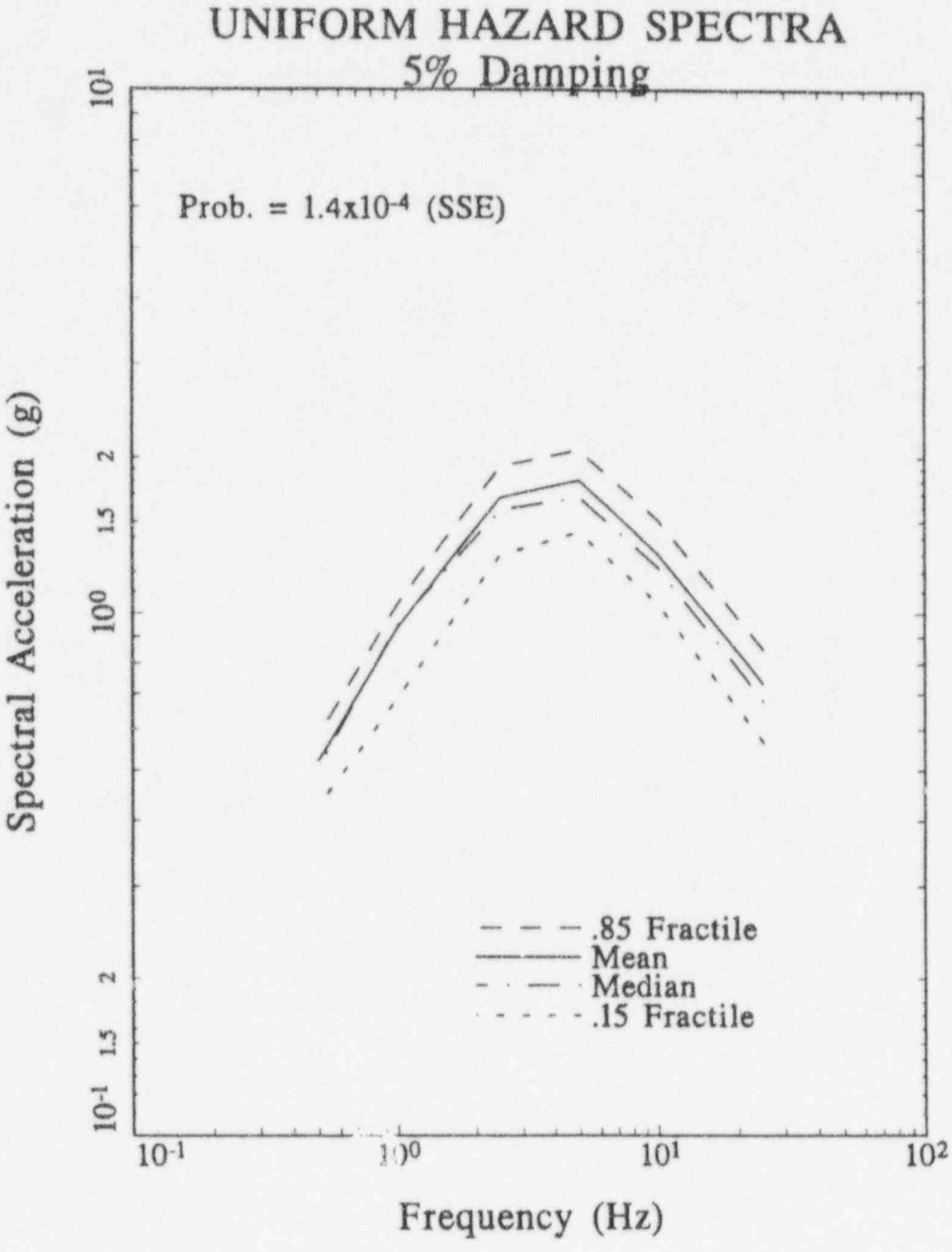


FIGURE 3.1-22
FRACTILES AND MEAN OF UNIFORM HAZARD SPECTRA, 1.7X10^{-6} (2XSSE) ANNUAL PROBABILITY OF EXCEEDENCE

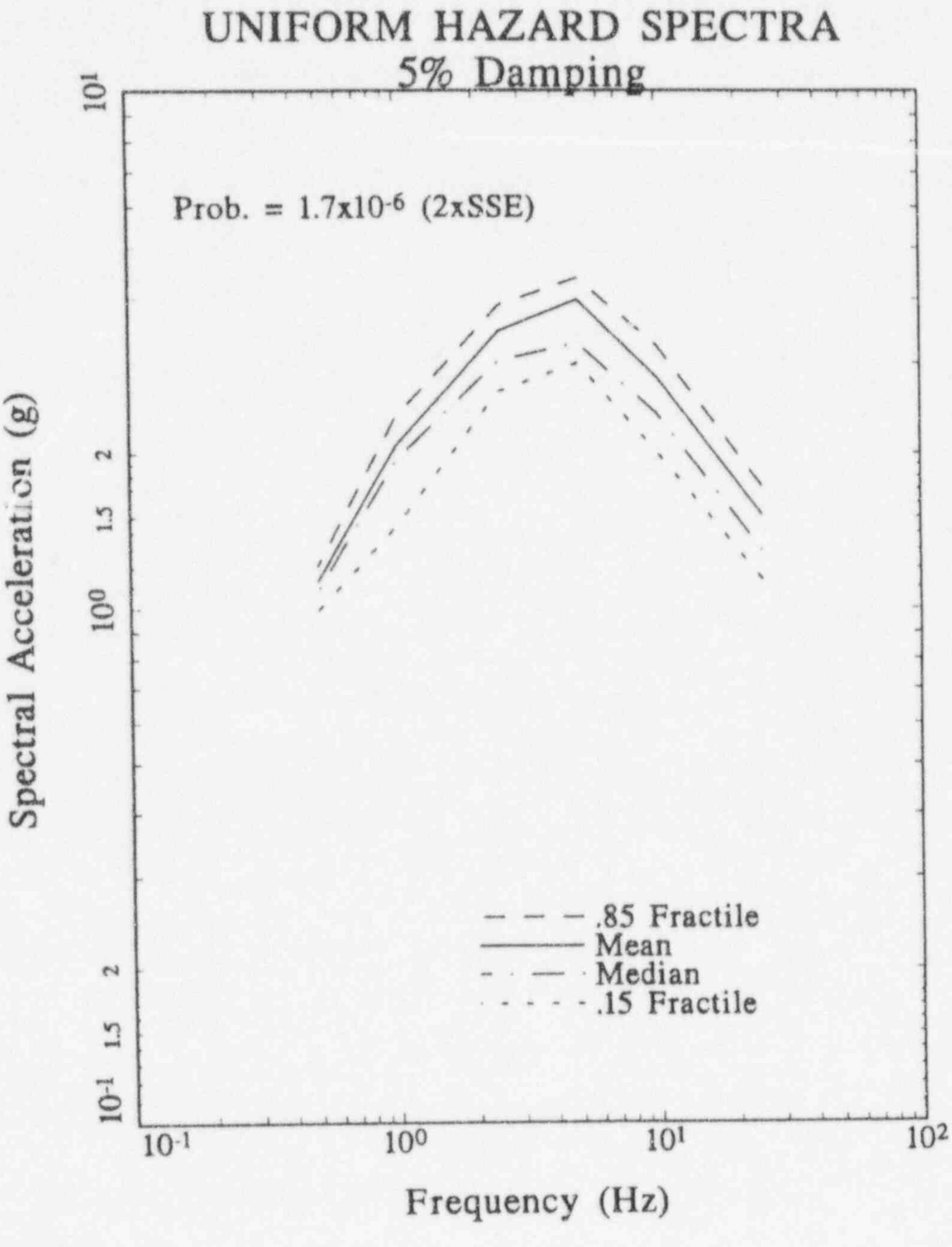


FIGURE 3.1-23
MEAN UNIFORM HORIZONTAL HAZARD SPECTRA

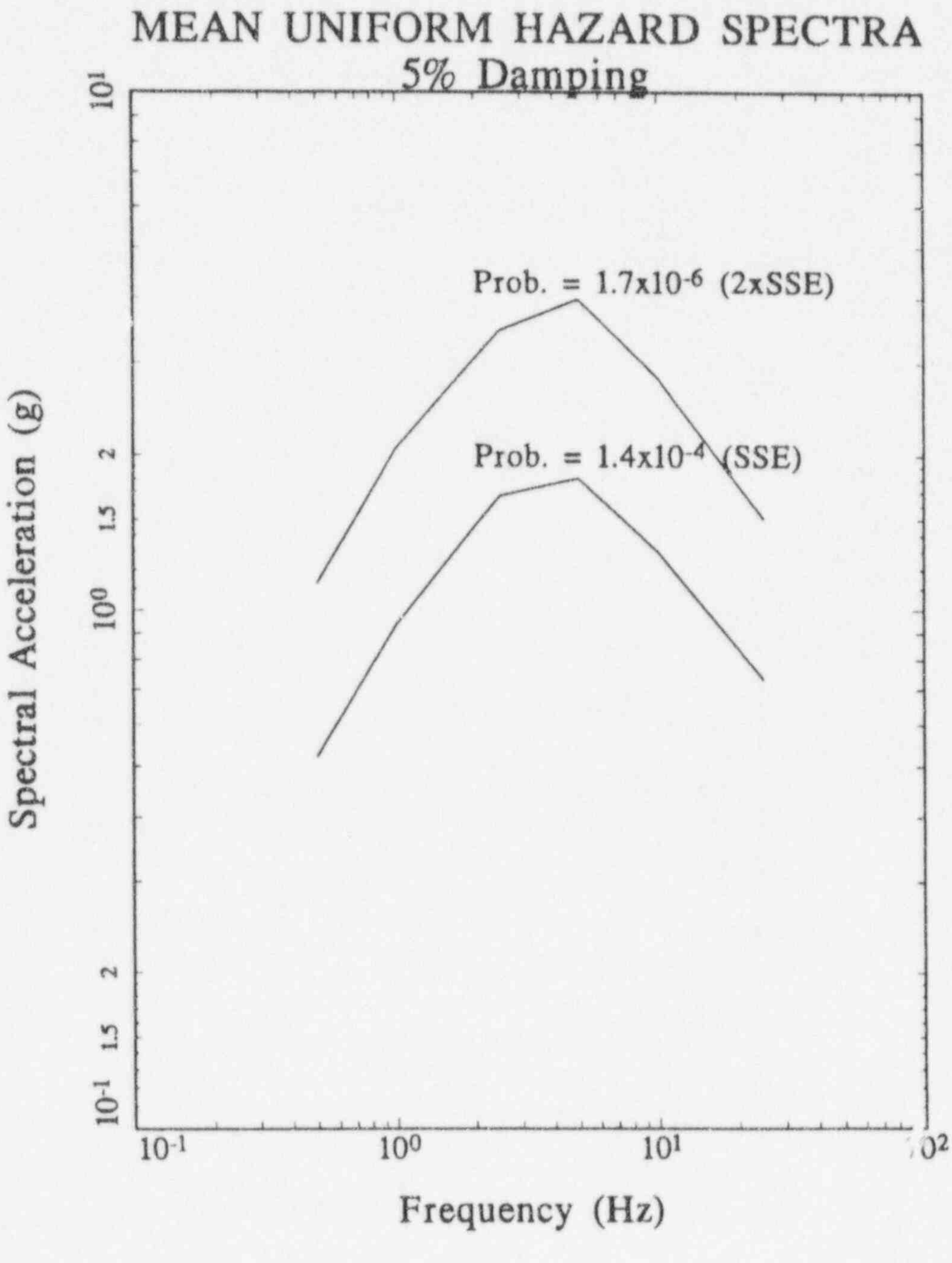


FIGURE 3.1-24
FRACTILES AND MEAN OF UNIFORM HAZARD SPECTRA FOR VERTICAL GROUND MOTIONS

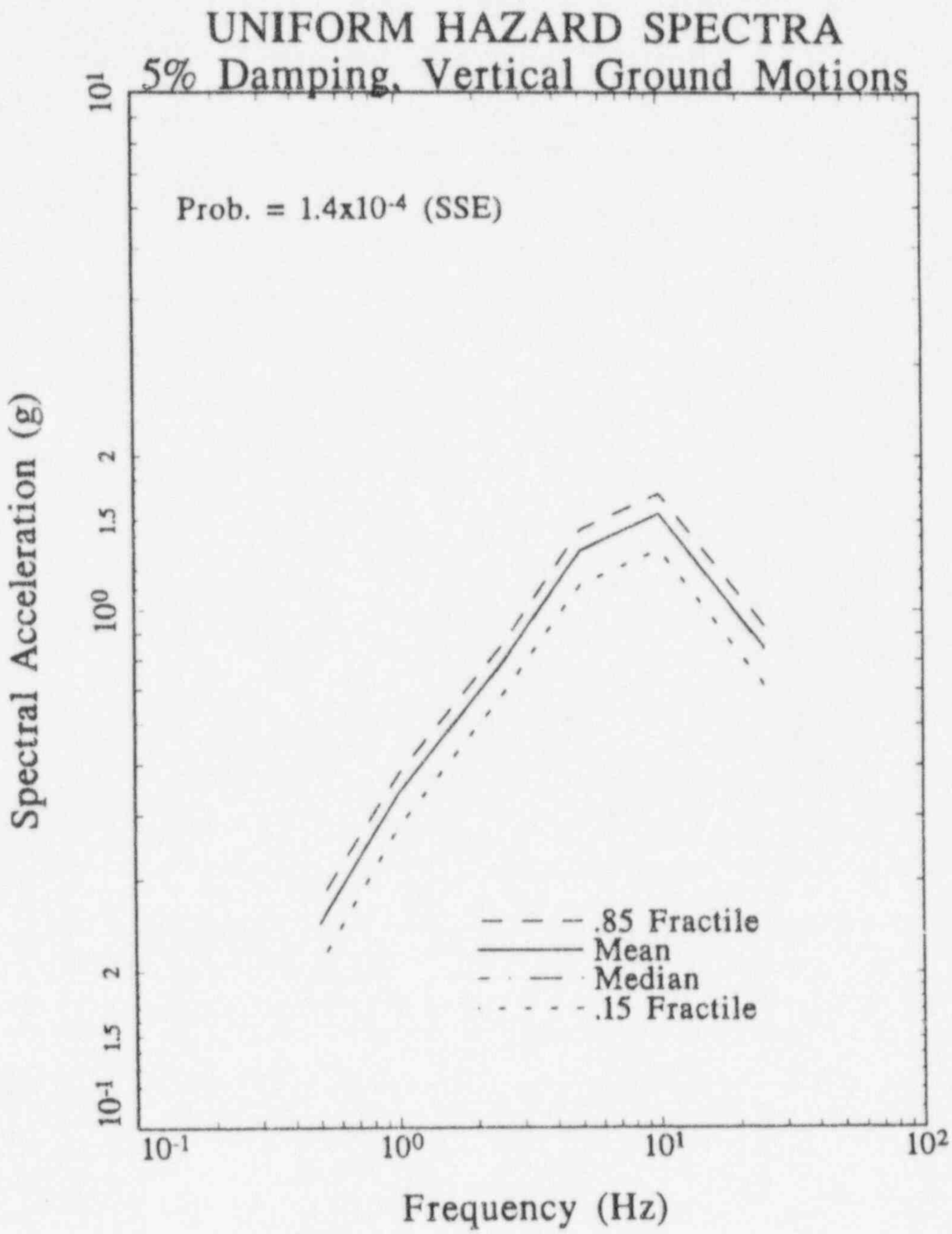


FIGURE 3.1-25
COMPARISON OF VERTICAL SPECTRAL SHAPES (NORMALIZED 1-10 HZ) FOR THE SSE AND 2xSSE LEVEL

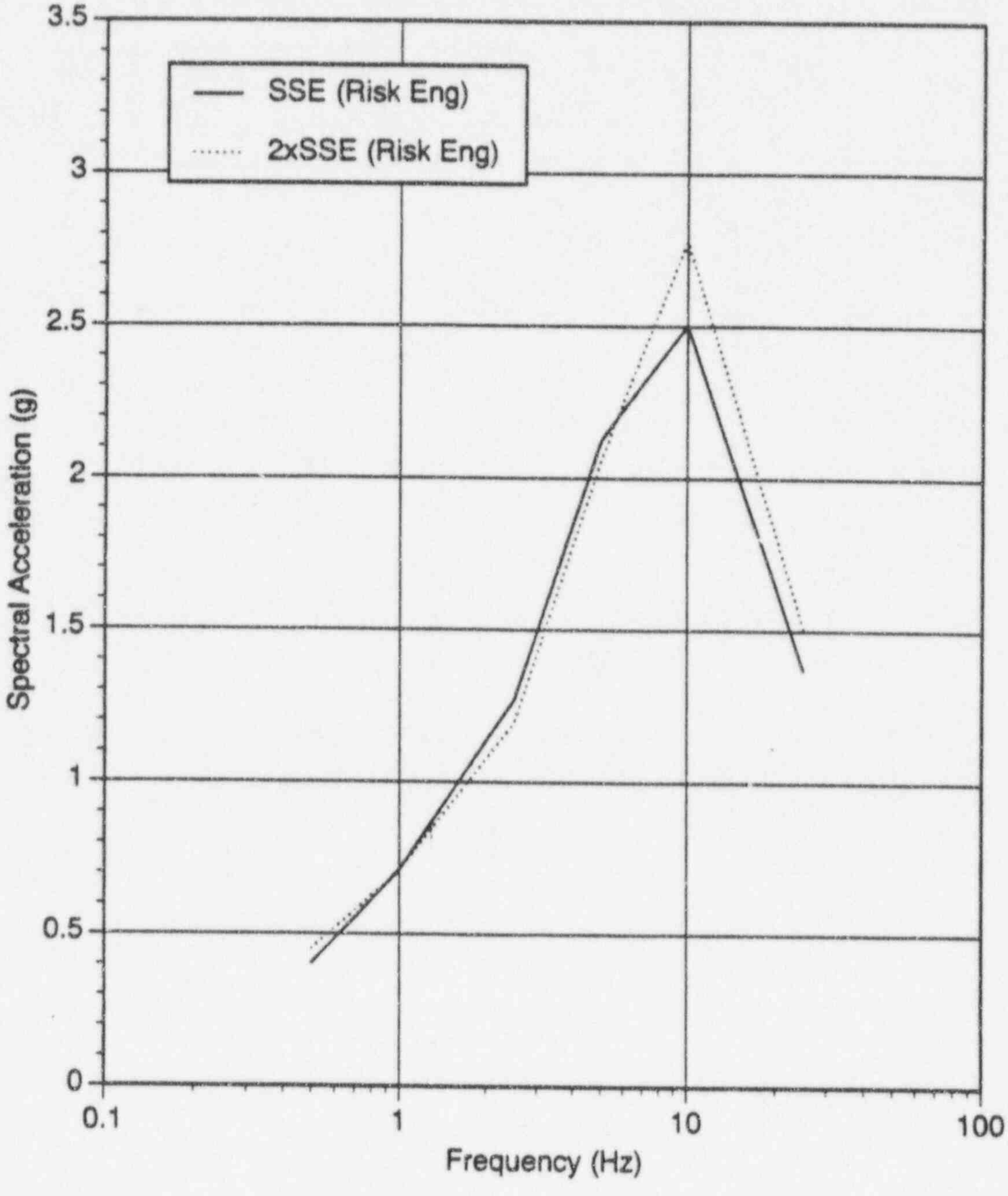


FIGURE 3.1-26
COMPARISONS OF STANDARD ERRORS FOR CAMPBELL (1990) AND SADIGH (1993) VERTICAL ATTENUATION EQUATIONS

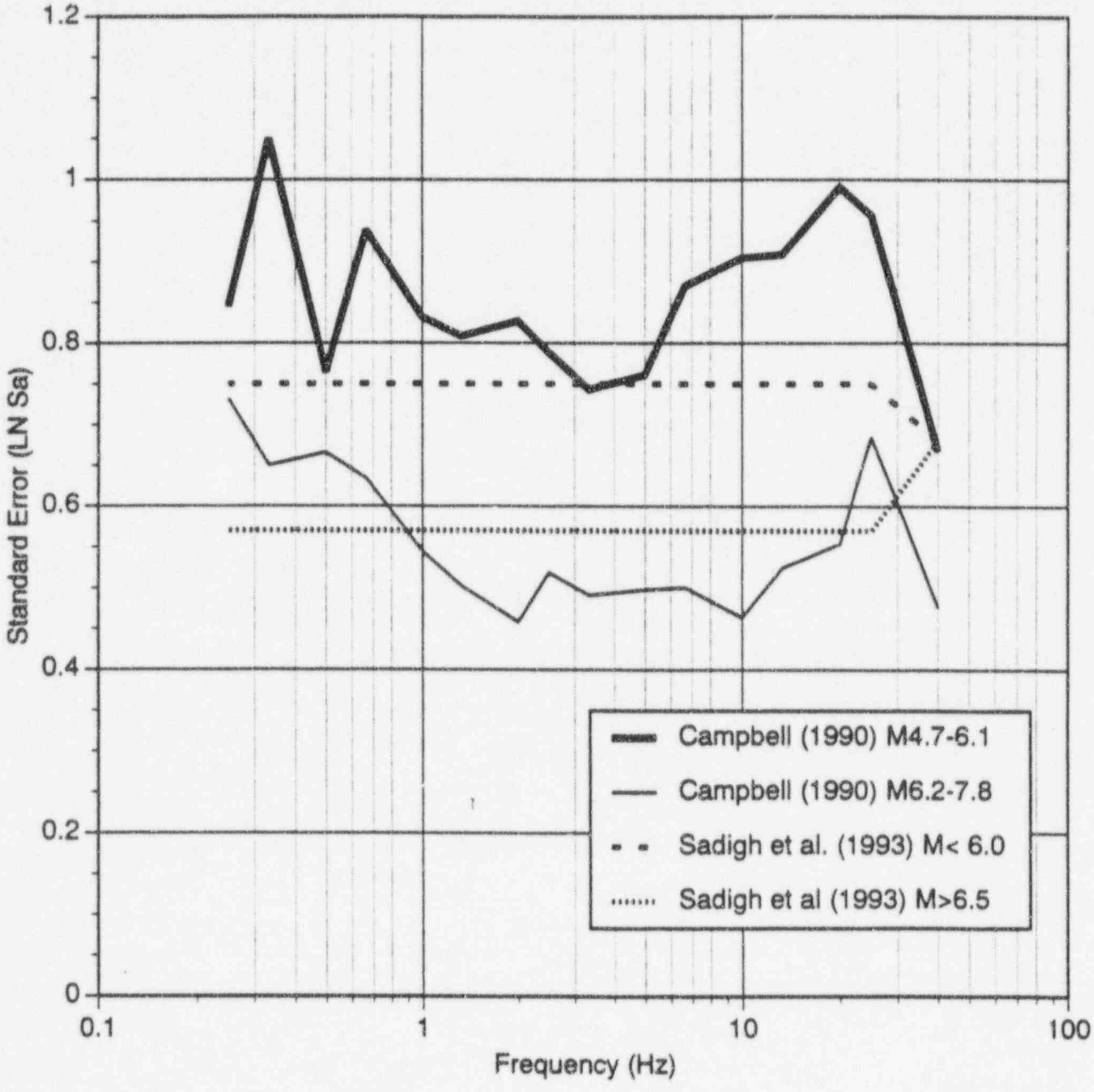


FIGURE 3.1-27
COMPARISON OF VERTICAL SPECTRAL SHAPES AT THE SSE LEVEL USING FULL HAZARD ANALYSIS AND SIMPLIFIED ANALYSIS INCLUDING ONLY SCOZD

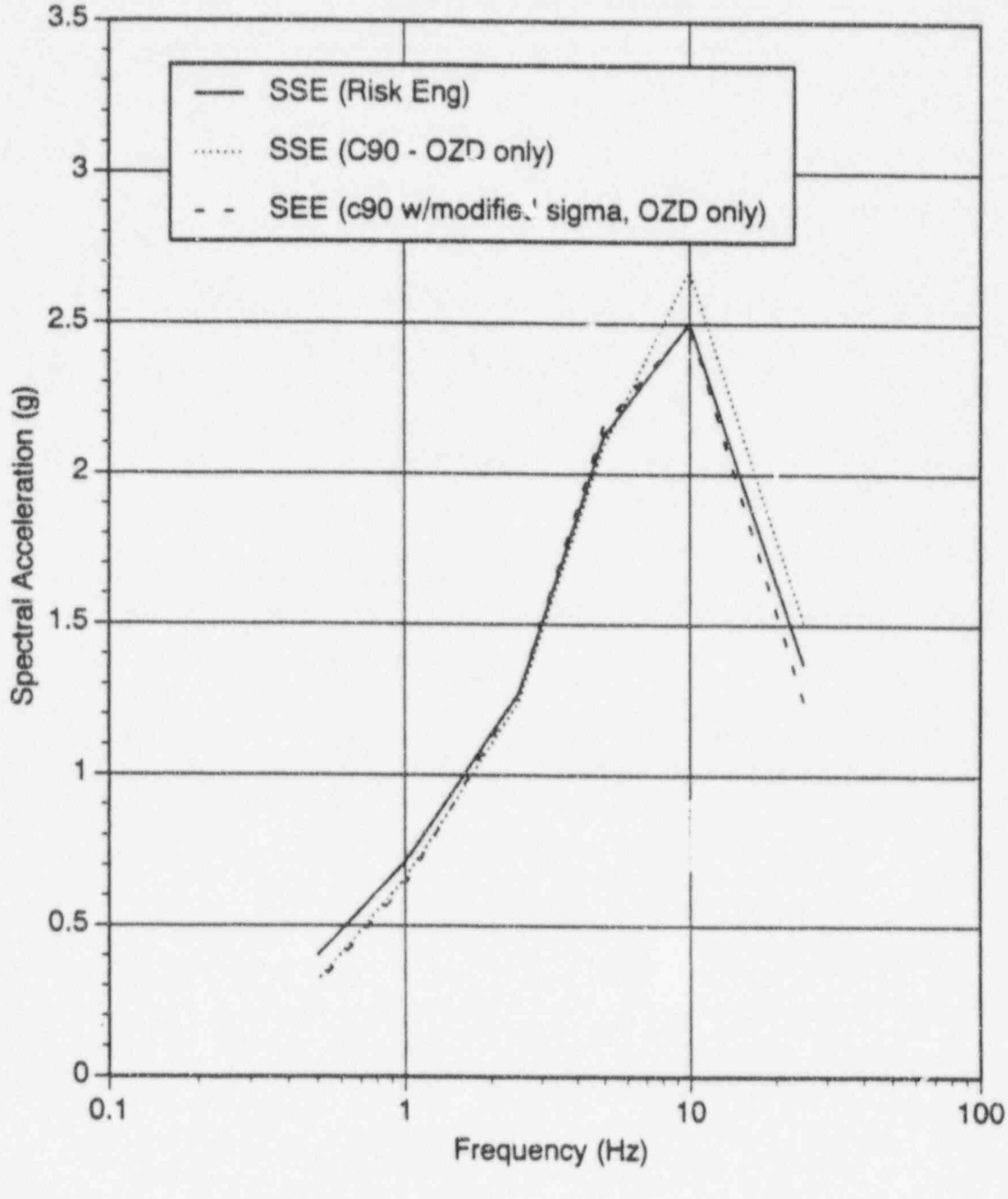


FIGURE 3.1-28
COMPARISON OF VERTICAL SPECTRAL SHAPES
AT THE SSE AND 2xSSE LEVELS

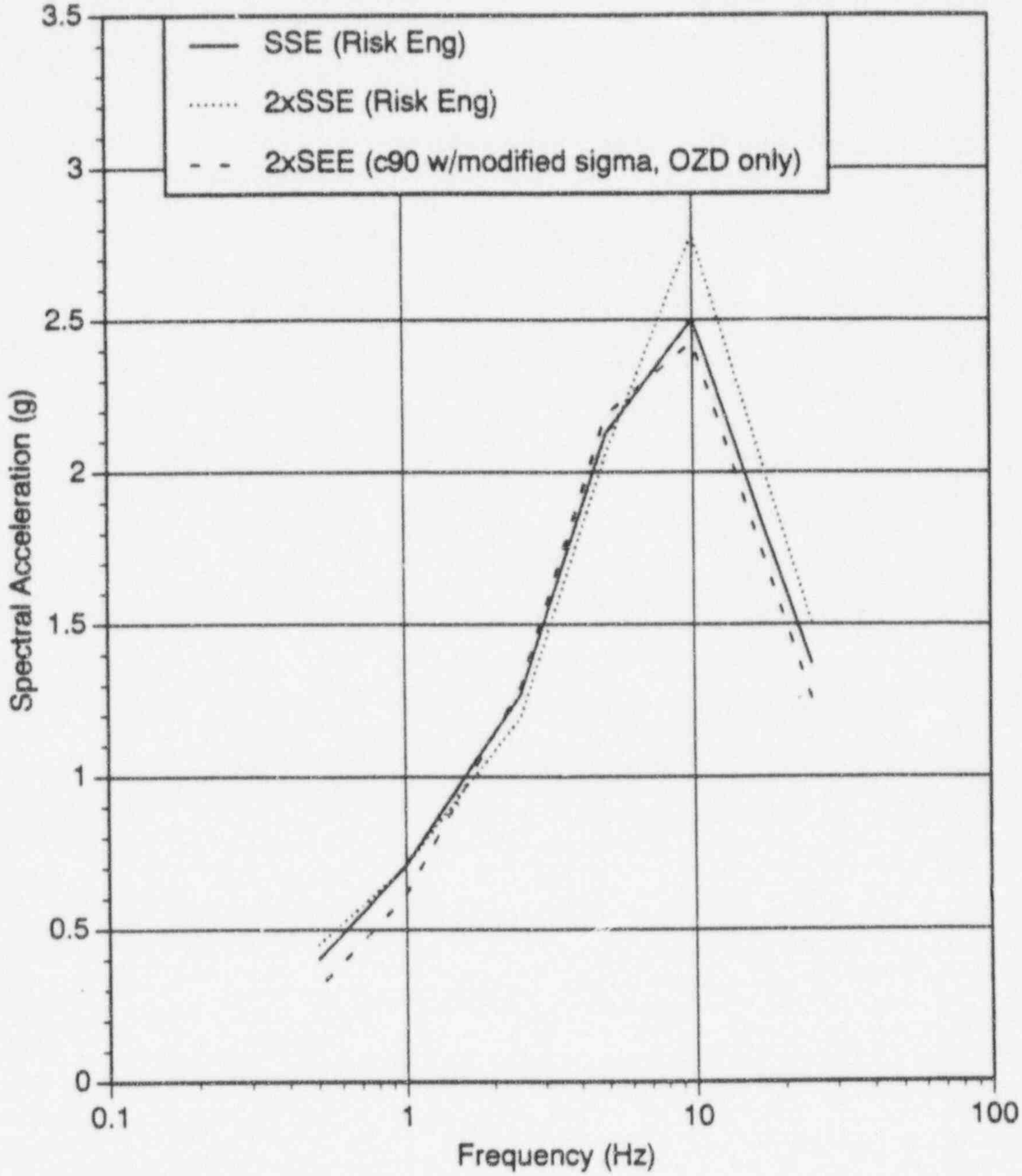
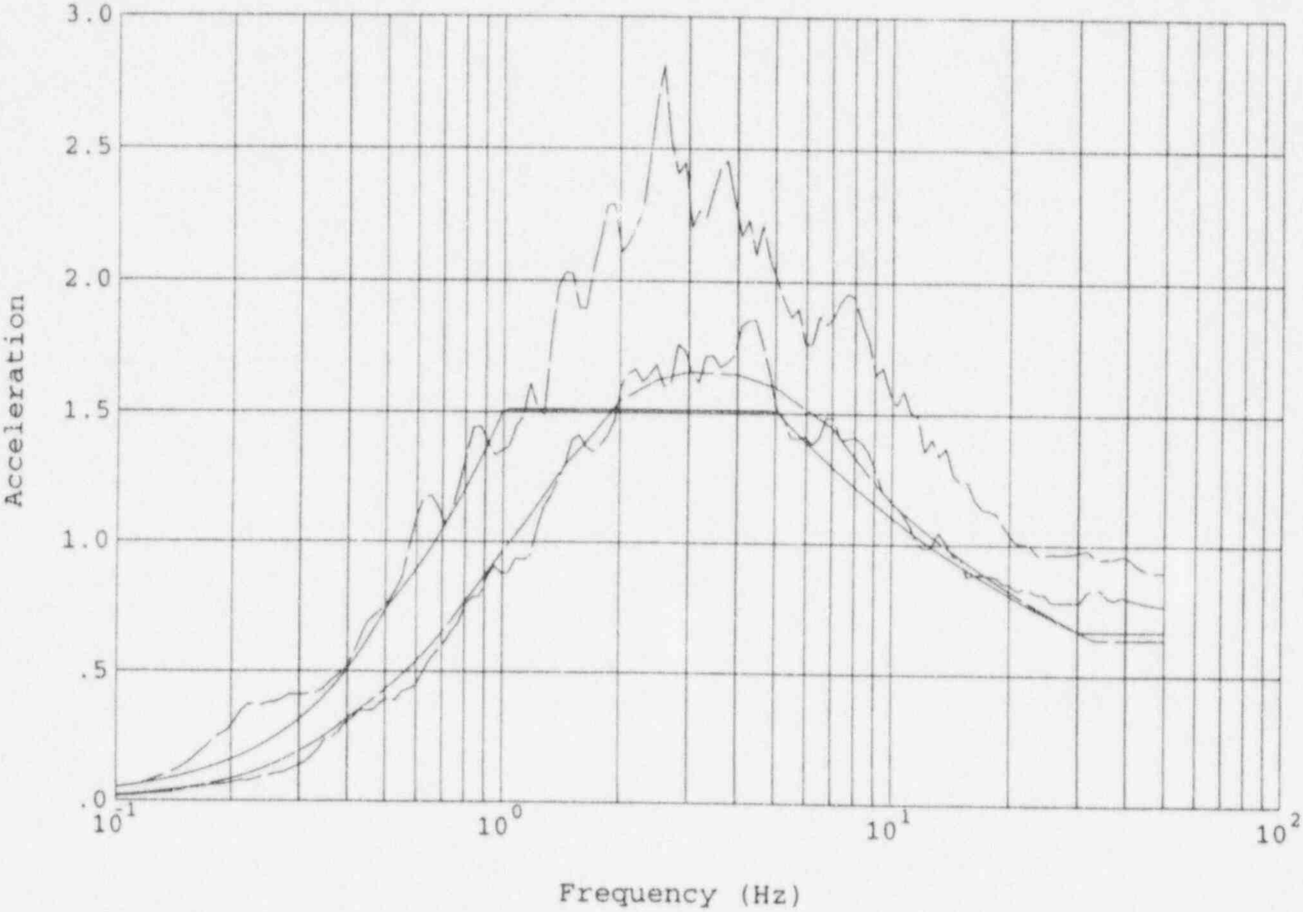


FIGURE 3.4-1
COMPARISON OF ENSEMBLE 50TH AND 84TH NEP SPECTRA AGAINST UNIFORM HAZARD SPECTRA, X-DIRECTION



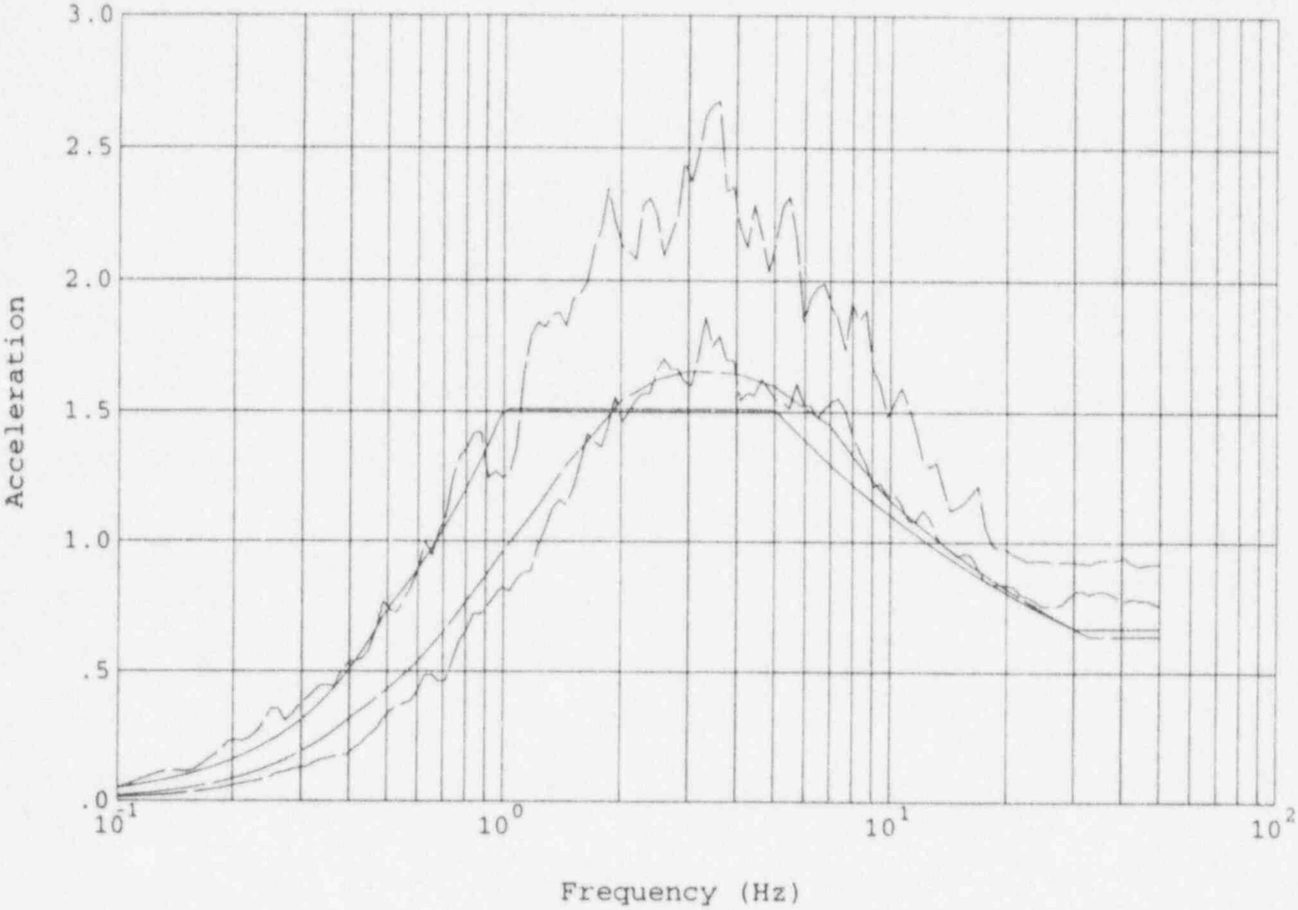
Legend:

DBE (HORIZONTAL) ——————
HORIZ UHS, SSE Level —— ——
Ensemble 50% (X) —— —— —
Ensemble 84% (X) —— —— —— –

Notes:

Acceleration in g's
Spectral acceleration at 5% damping

FIGURE 3.4-2
COMPARISON OF ENSEMBLE 50TH AND 84TH NEP SPECTRA AGAINST UNIFORM HAZARD SPECTRA, Y-DIRECTION



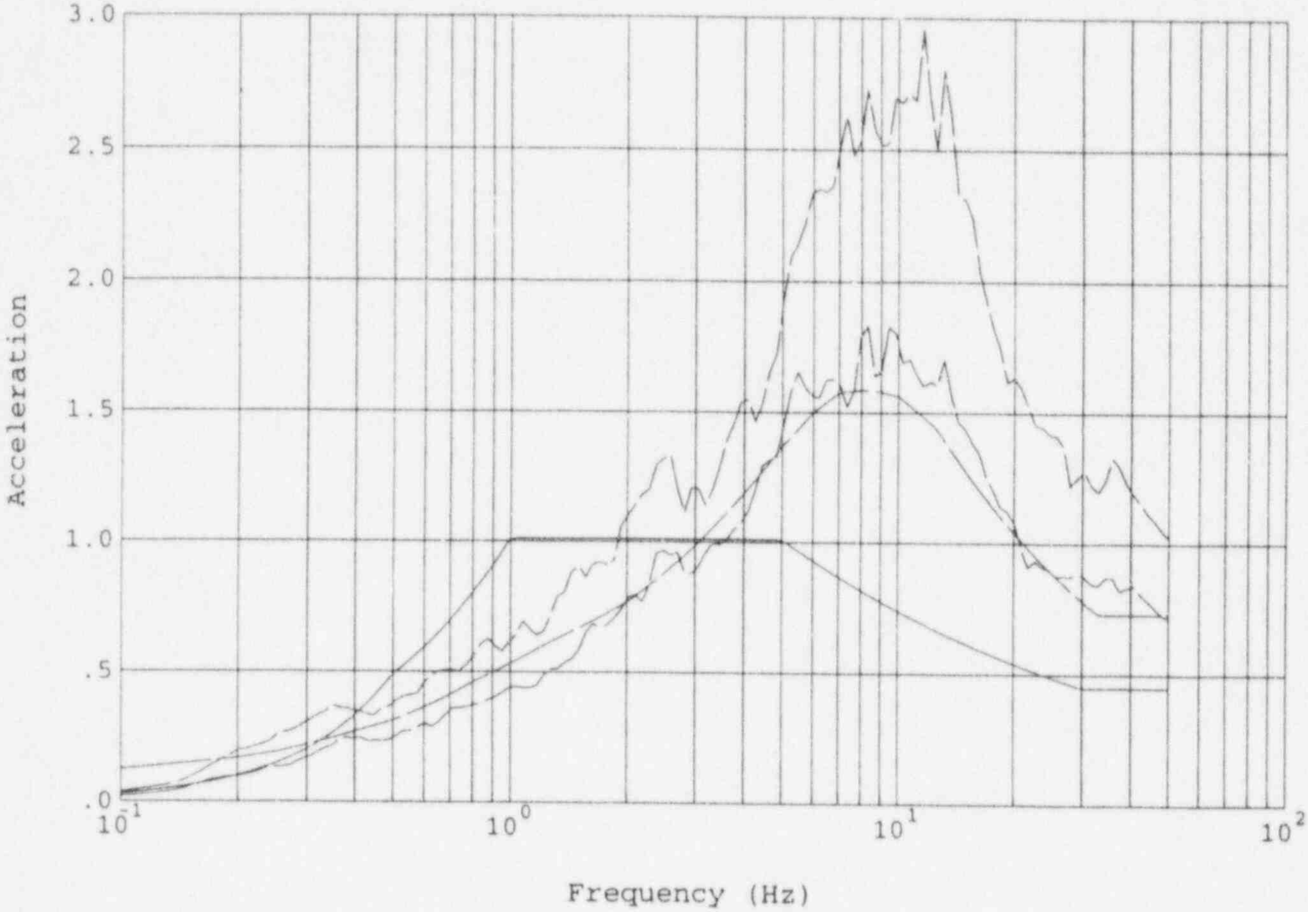
Legend:

DBE (HORIZONTAL)
HORIZ UHS, SSE Level
Ensemble 50% (Y)
Ensemble 84% (Y)

Notes:

Acceleration in g's
Spectral acceleration at 5% damping

FIGURE 3.4-3
COMPARISON OF ENSEMBLE 50TH AND 84TH NEP SPECTRA AGAINST UNIFORM HAZARD SPECTRA, Z-DIRECTION



Legend:

DBE (VERTICAL)
VERT UHS, SSE Level
Ensemble 50% (Vert)
Ensemble 84% (Vert)

Notes:

Acceleration in g's
Spectral acceleration at 5% damping

FIGURE 3.4-4
COMPARISON OF NORMALIZED PRELIMINARY AND FINAL UHS (HORIZONTAL)

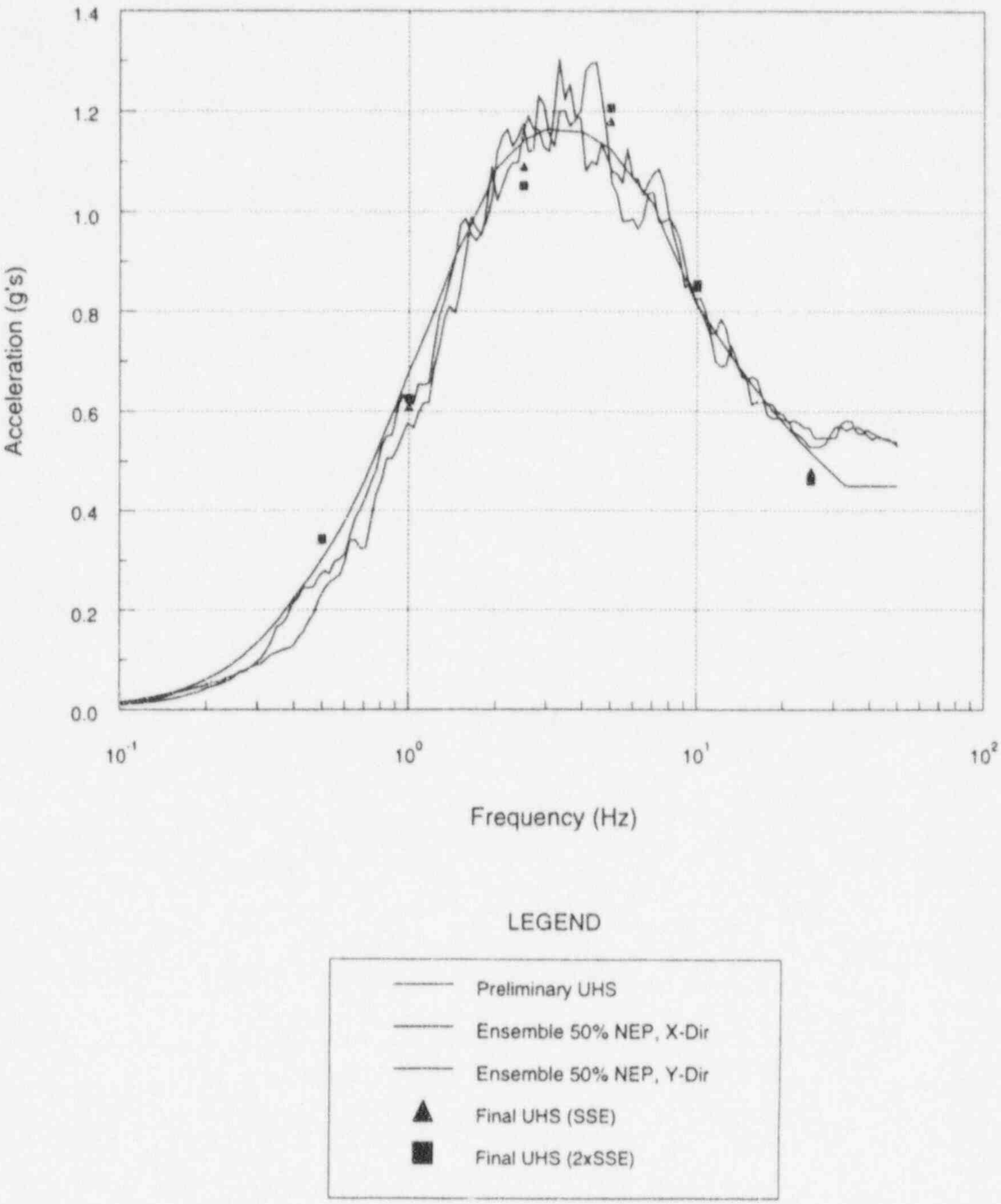


FIGURE 3.4-5
COMPARISON OF NORMALIZED PRELIMINARY AND FINAL UHS (VERTICAL)

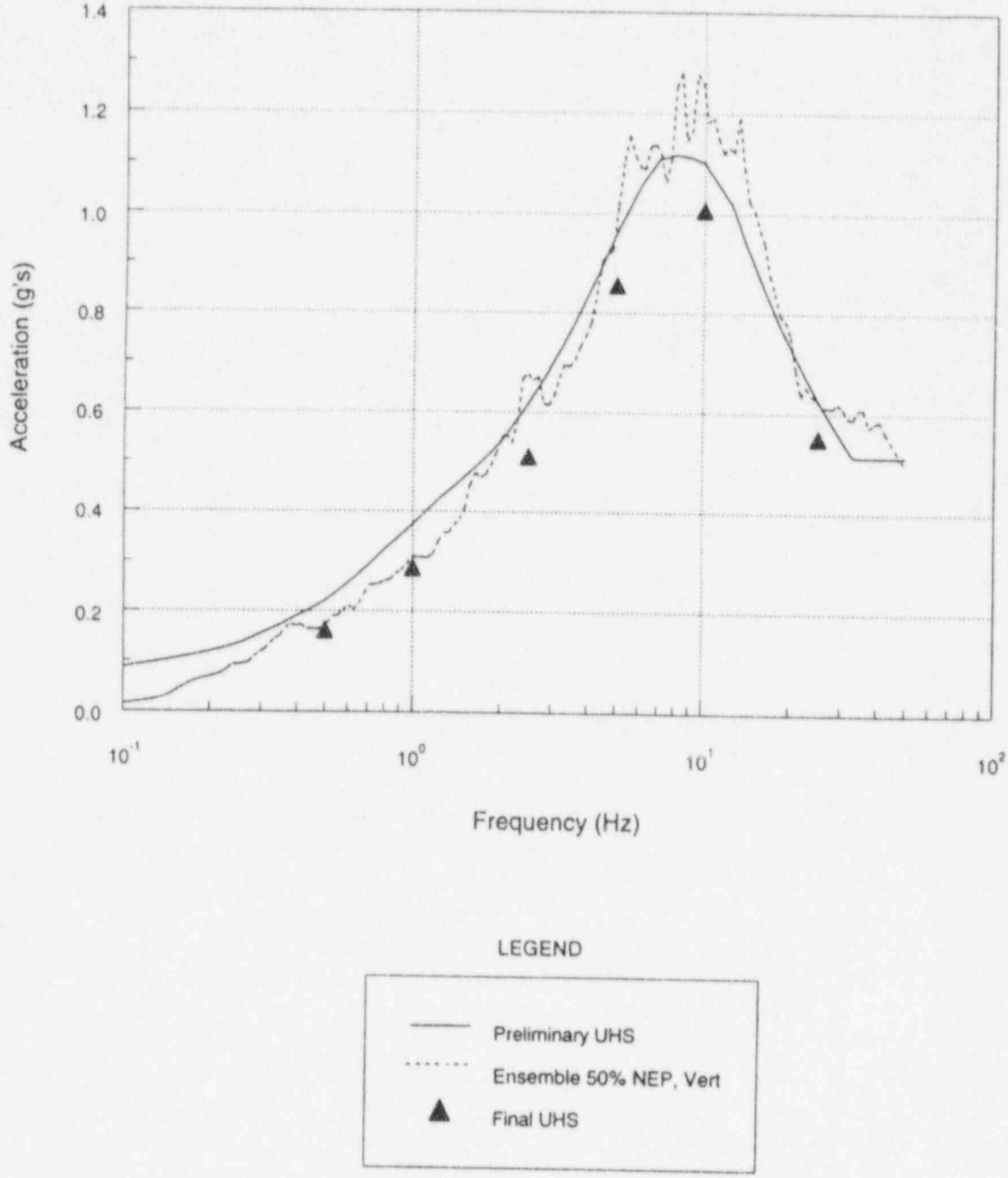


FIGURE 3.4-6
SOIL PROFILE UNDER VARIOUS STRUCTURE

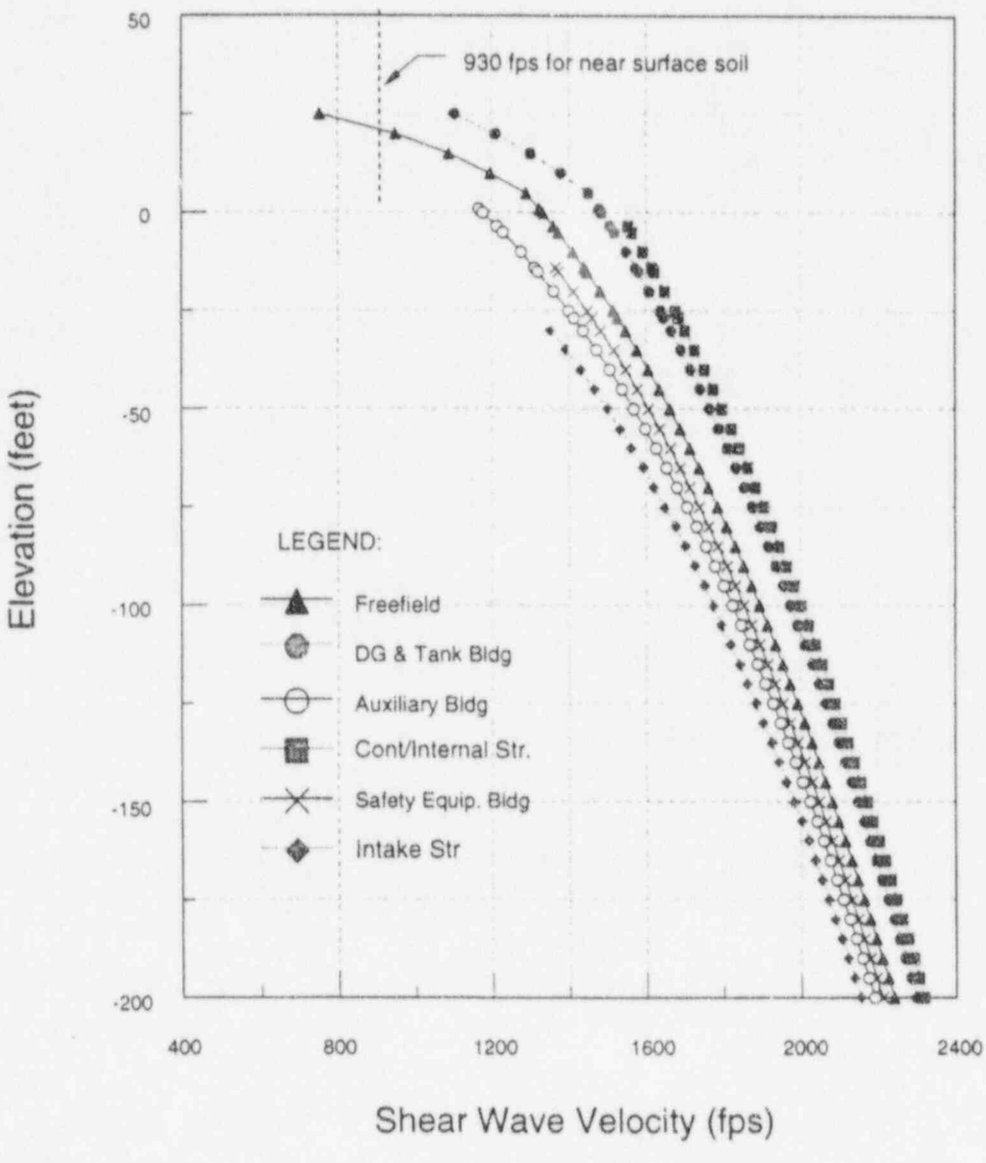
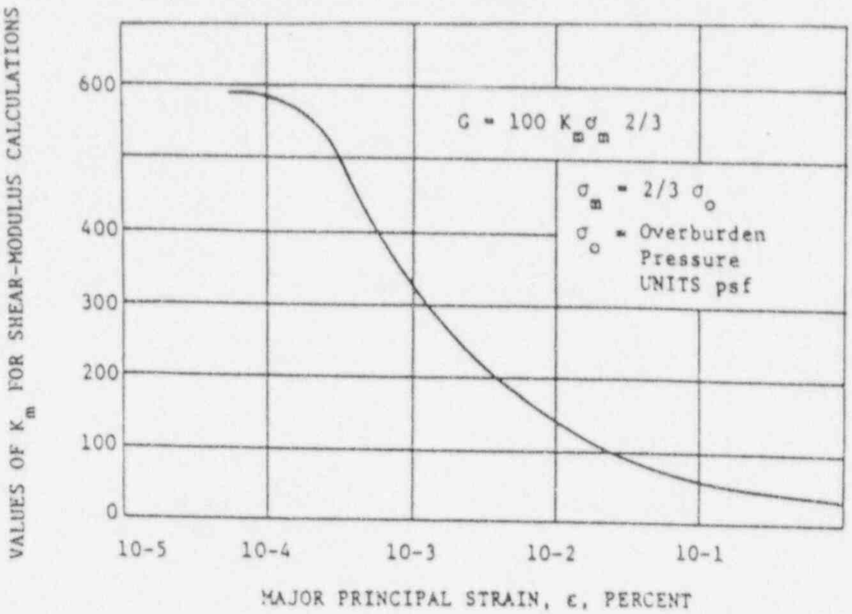
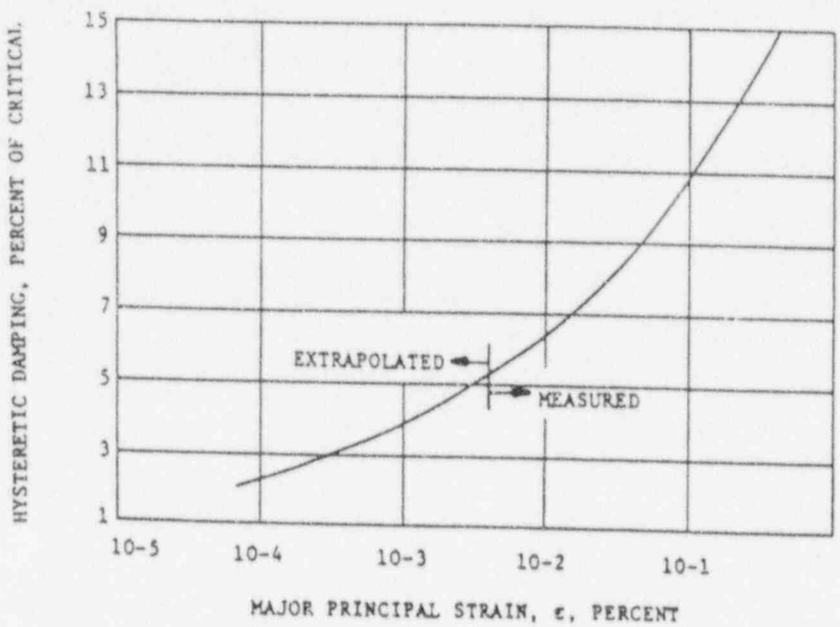


FIGURE 3.4-7
DYNAMIC SOIL PROPERTIES FROM FSAR





Updated

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
MODULUS AND DAMPING vs. STRAIN SAN MATEO FORMATION SAND

FIGURE 3.4-8
SHEAR MODULUS DEGRADATION CURVE FOR SAN MATEO SAND

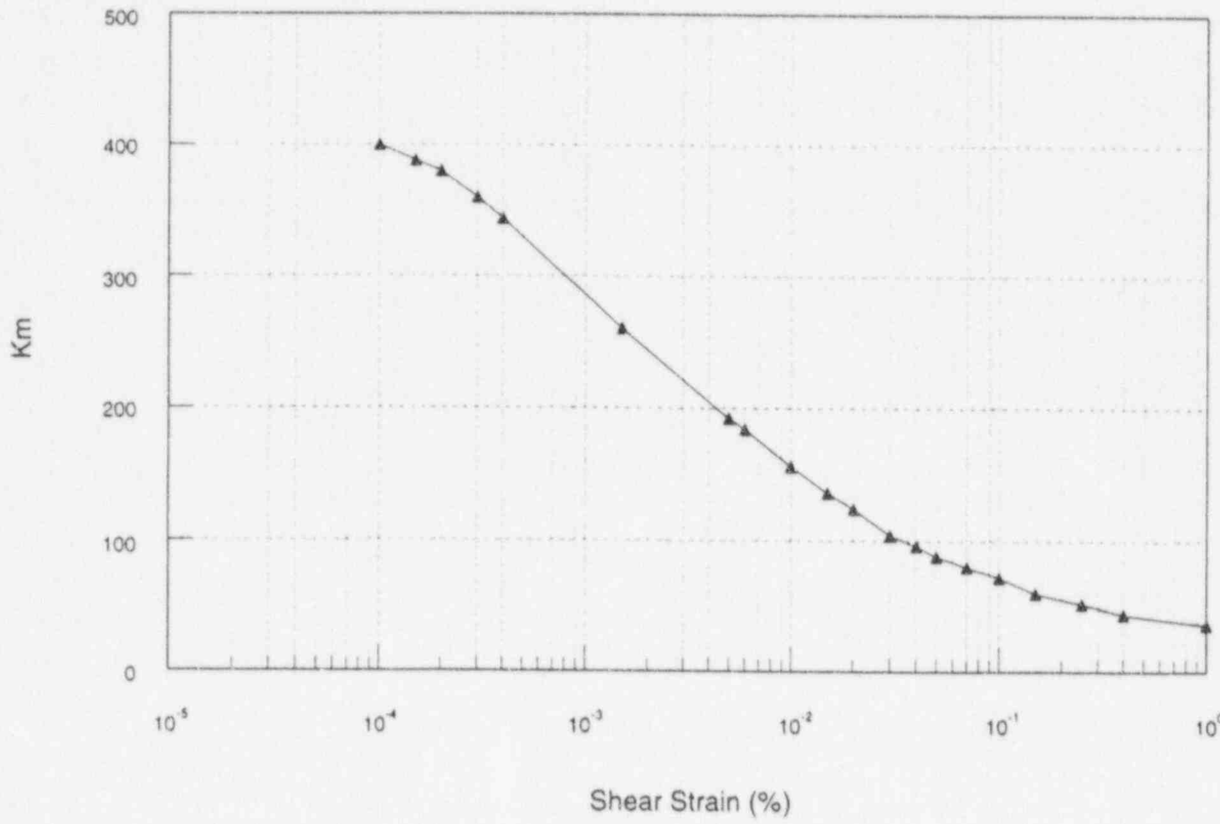
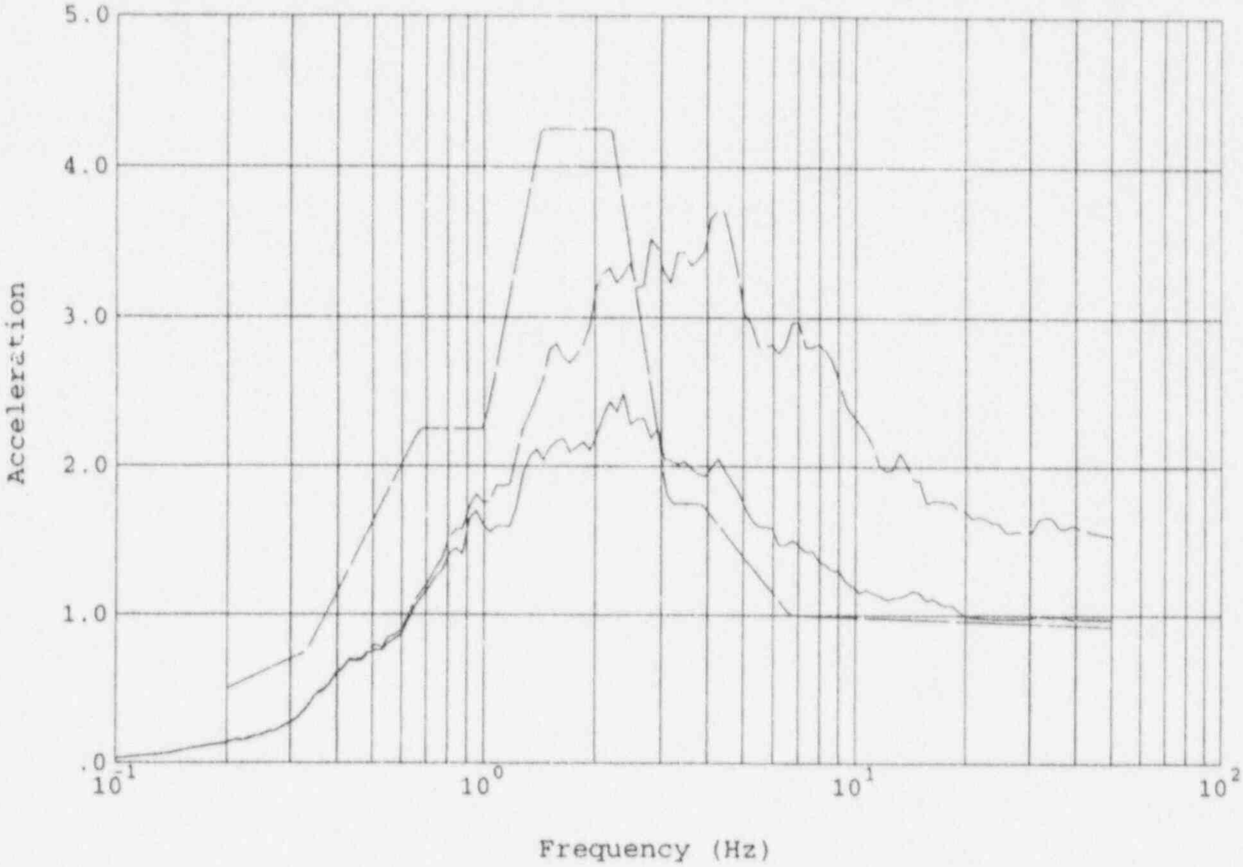


FIGURE 3.4-9
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
AUXILIARY BUILDING - TOP OF BASEMAT @ EL. 9'-0"



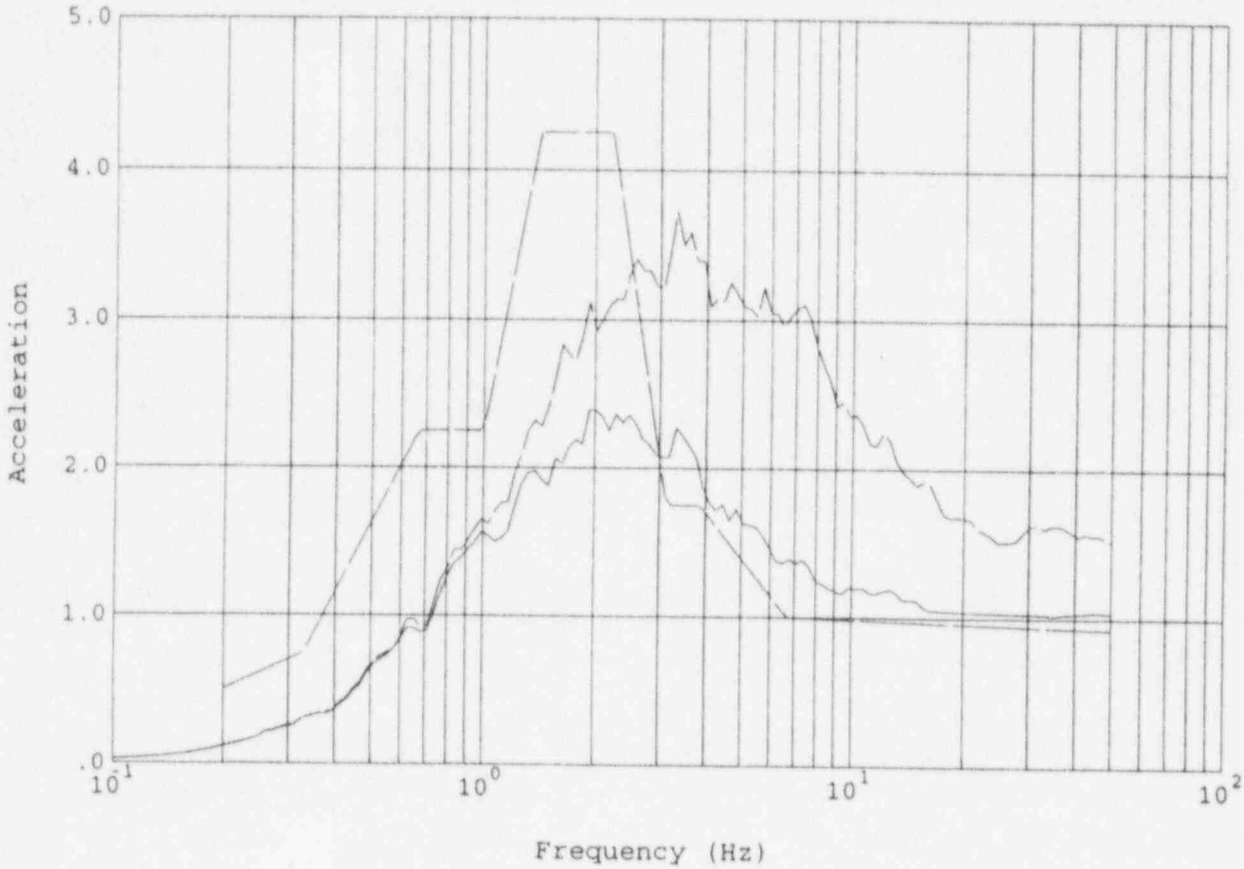
Legend:

Probabilistic Median ————
FSAR SSE Level —— ——
Freefield 50% NEP —— —— —

Notes:

N-S (X) Translation
5% damped spectral acceleration
Acceleration in g's

FIGURE 3.4-10
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
AUXILIARY BUILDING - TOP OF BASEMAT @ EL. 9'-0"



Legend:

Probabilistic Median ————

FSAR SSE Level —— ——

Freefield 50% NEP —— —— —

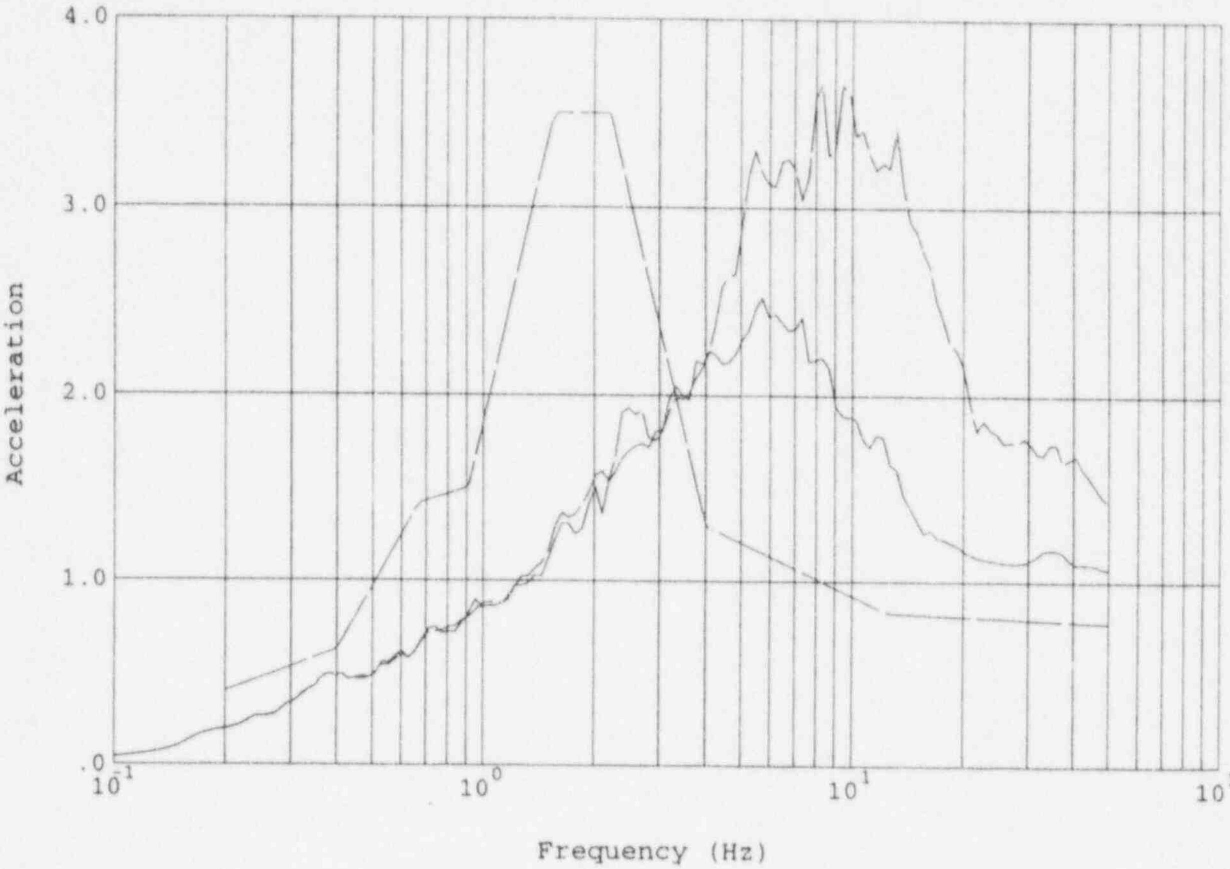
Notes:

E-W (Y) Translation

5% damped spectral acceleration

Acceleration in g's

FIGURE 3.4-11
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
AUXILIARY BUILDING - TOP OF BASEMAT @ EL. 9'-0"



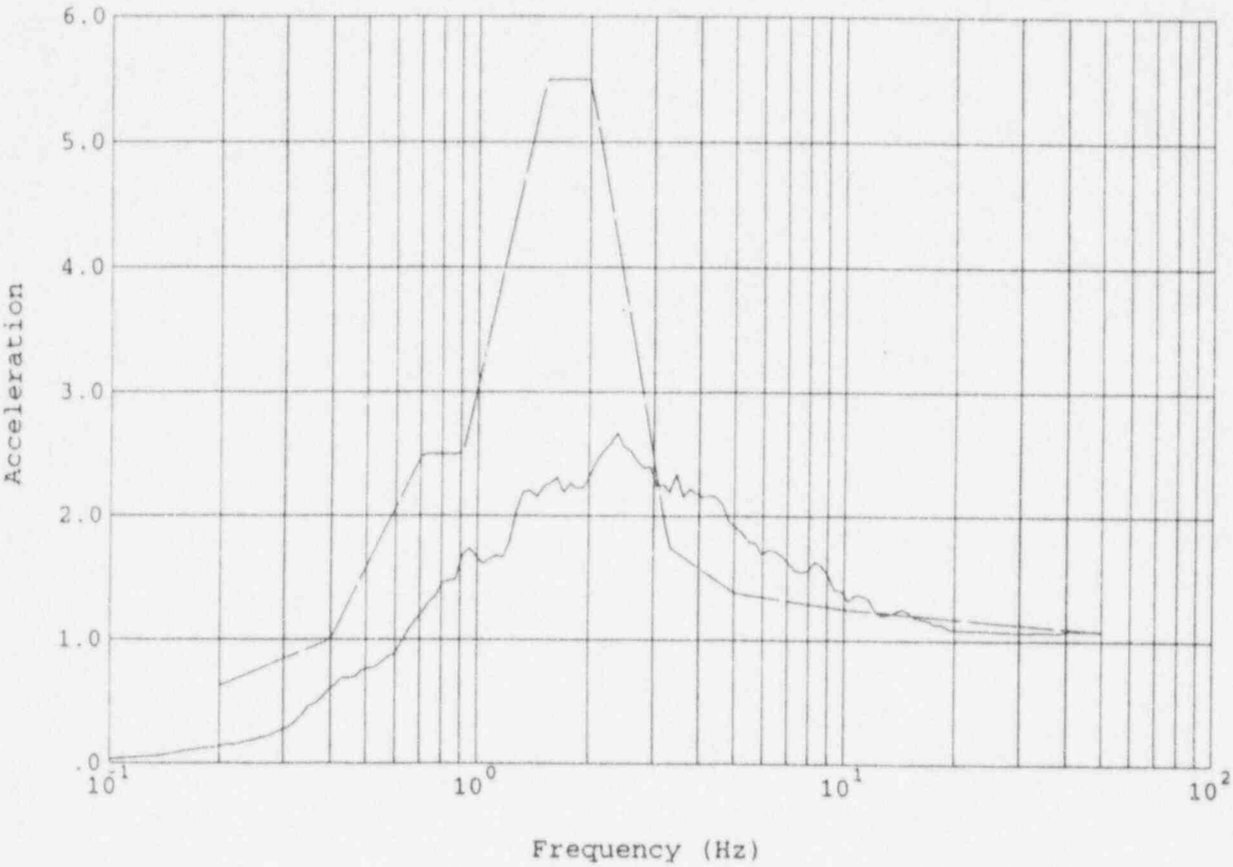
Legend:

Probabilistic Median ___________
FSAR SSE Level _____ _____
Freefield 50% NEP ____ ____ __

Notes:

Vertical Translation
5% damped spectral acceleration
Acceleration in g's

FIGURE 3.4-12
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
AUXILIARY BUILDING - EL. 63'-6"



Legend:

Probabilistic Median ____________

FSAR SSE Level ______ ______

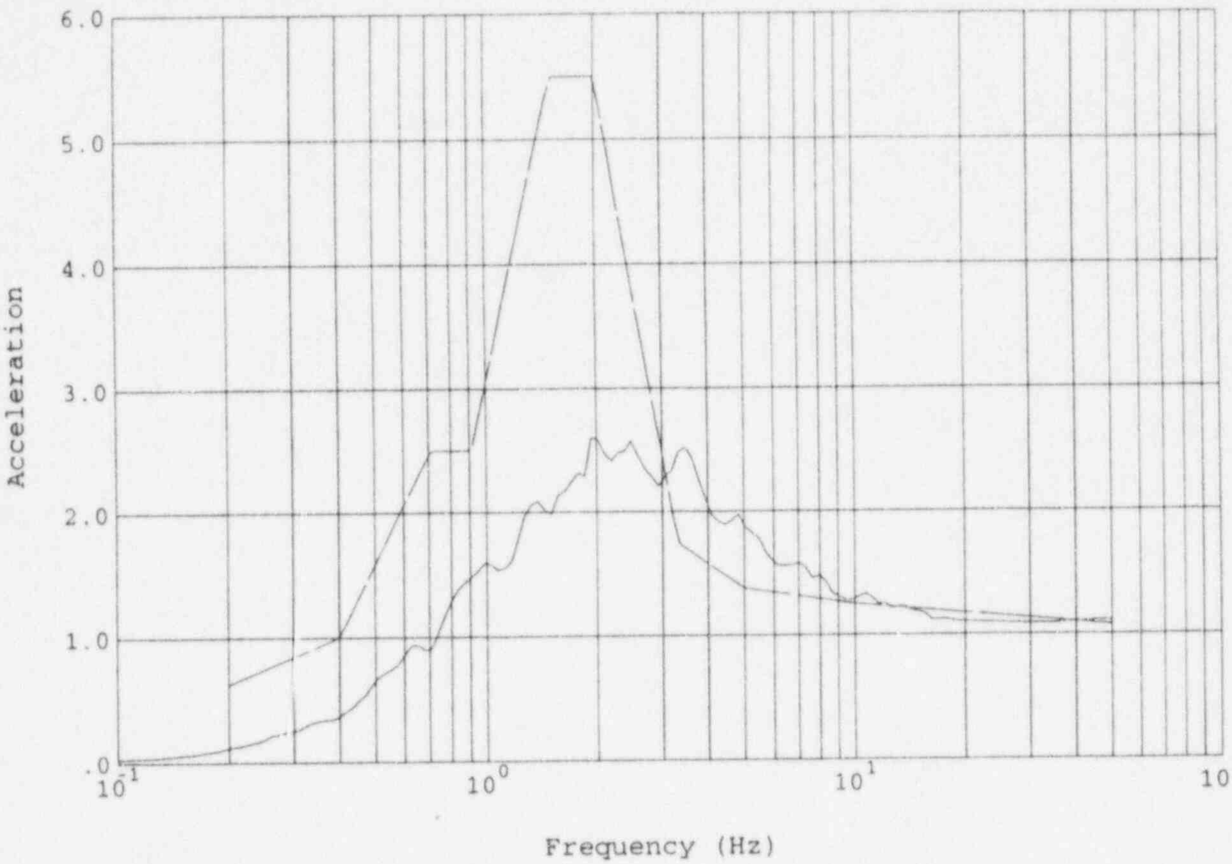
Notes:

N-S (X) Translation

5% damped spectral acceleration

Acceleration in g's

FIGURE 3.4-13
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
AUXILIARY BUILDING - EL. 63'-6"



Legend:

Probabilistic Median ————

FSAR SSE Level —— ——

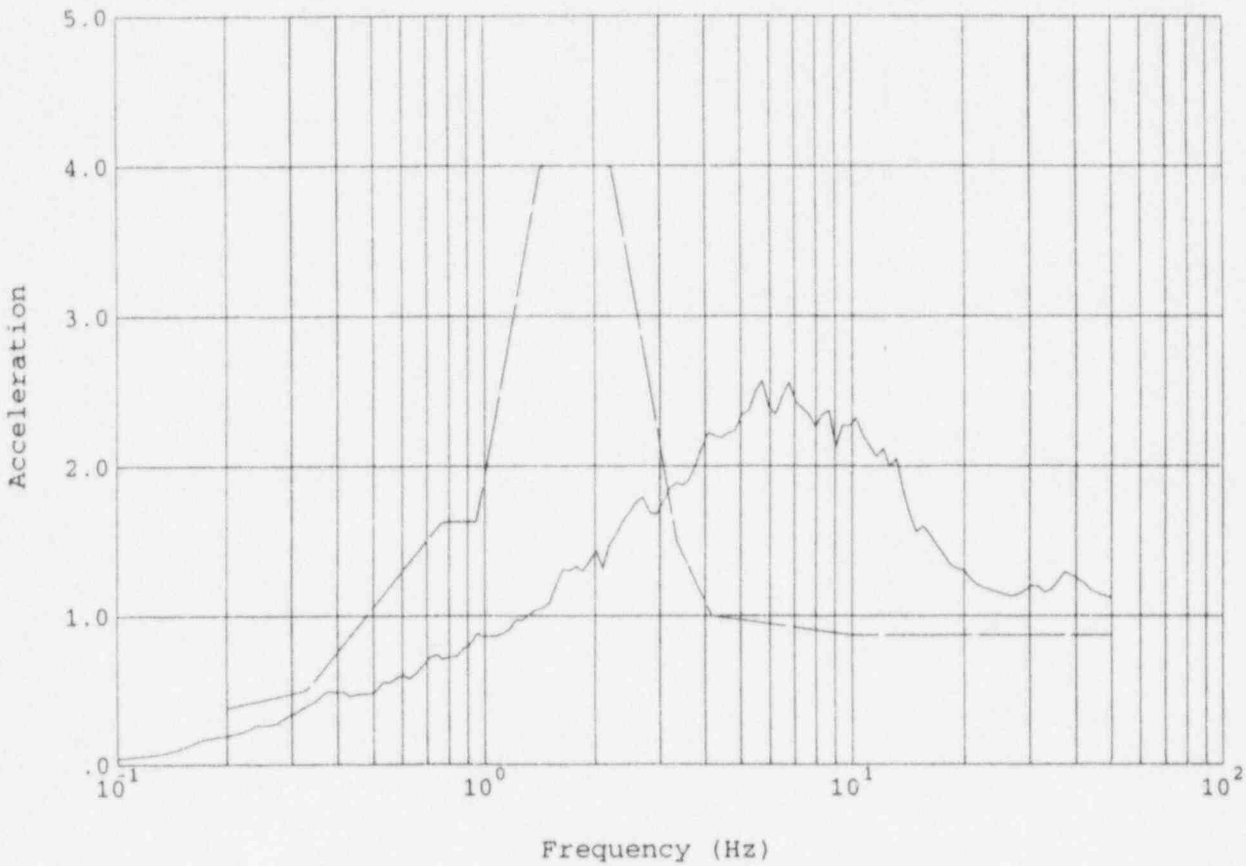
Notes:

E-W (Y) Translation

5% damped spectral acceleration

Acceleration in g's

FIGURE 3.4-14
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
AUXILIARY BUILDING - EL. 63'-6"



Legend:

Probabilistic Median

FSAR SSE Level

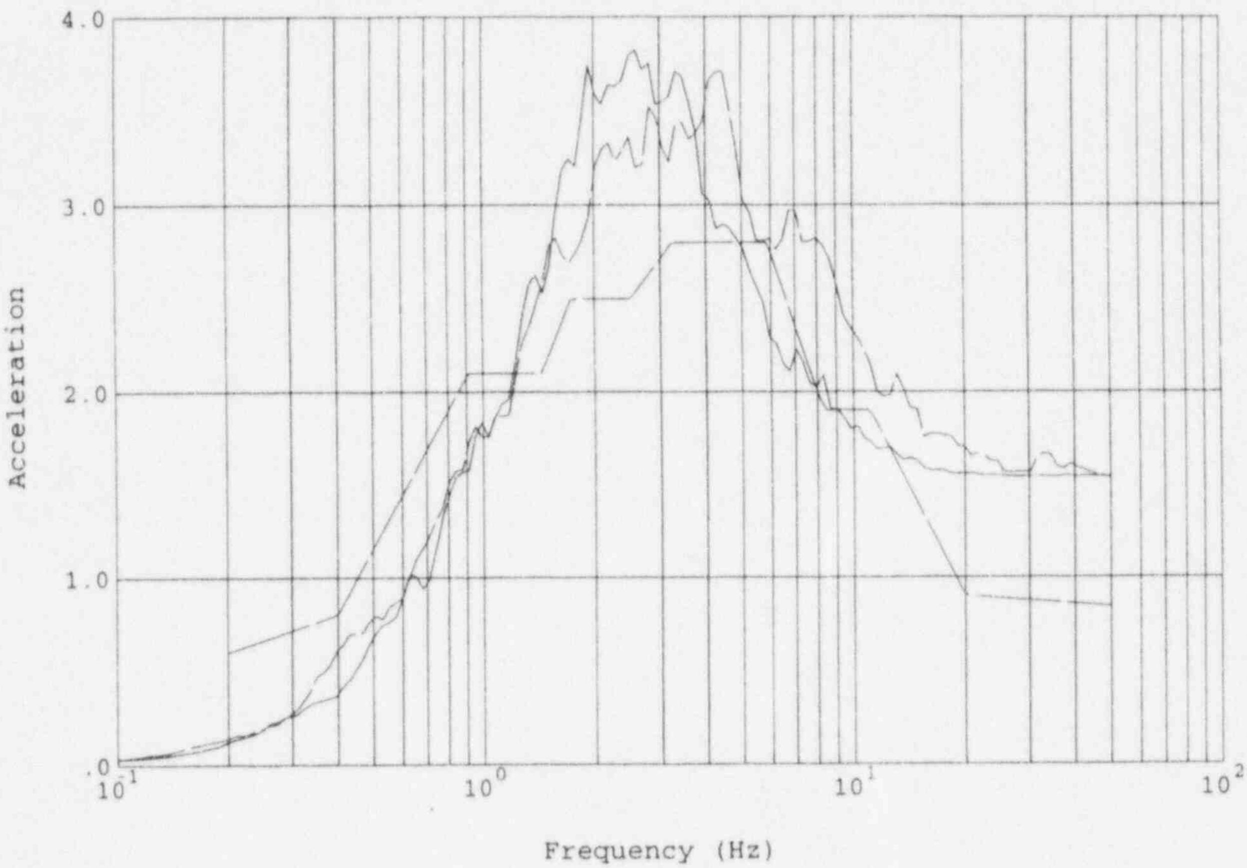
Notes:

Vertical Translation

5% damped spectral acceleration

Acceleration in g's

FIGURE 3.4-15
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
DIESEL GENERATOR BUILDING FOUNDATION @ EL. 27'-9"



Legend:

Probabilistic Median ————————

FSAR SSE Level ———— ————

Freefield 50% NEP ———— ———— ——

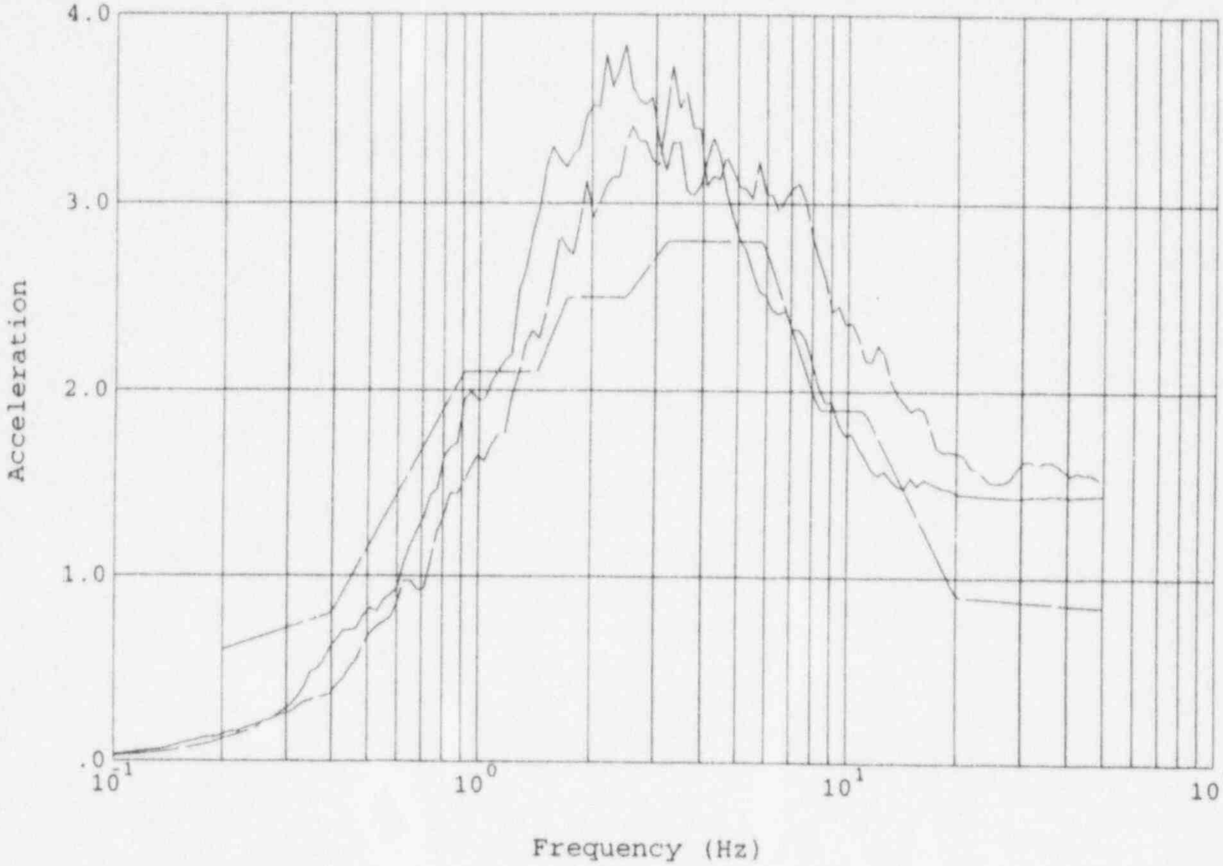
Notes:

E-W (X) Translation

5% damped spectral acceleration

Acceleration in g's

FIGURE 3.4-16
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
DIESEL GENERATOR BUILDING FOUNDATION @ EL. 27'-9"



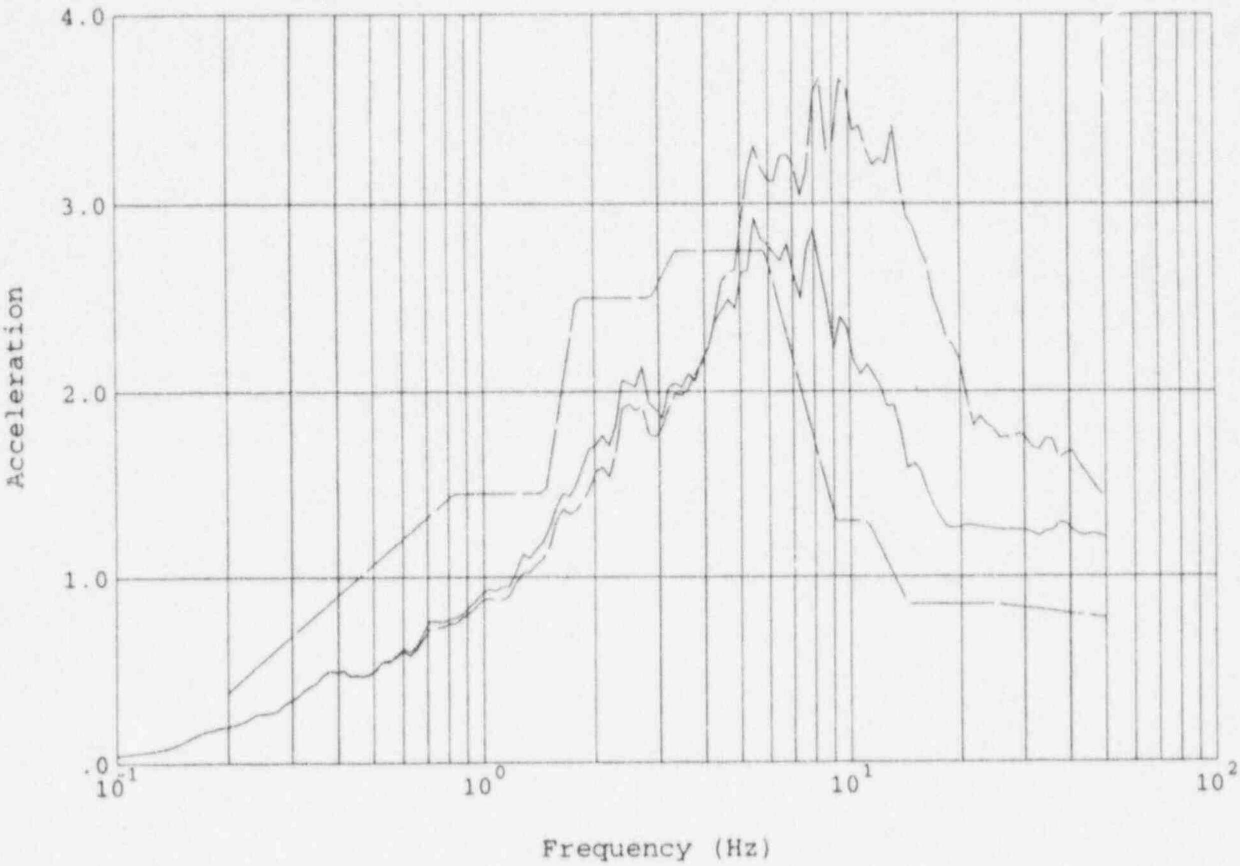
Legend:

Probabilistic Median ————————
FSAR SSE Level ——— ———
Freefield 50% NEP ——— ——— ——

Notes:

N-S (Y) Translation
5% damped spectral acceleration
Acceleration in g's

FIGURE 3.4-17
COMPARISON OF 2xSSE PROBABILISTIC VS. FSAR ANALYSIS
DIESEL GENERATOR BUILDING FOUNDATION @ EL. 27'-9"



Legend:

Probabilistic Median

FSAR SSE Level

Freefield 50% NEP

Notes:

Vertical (Z) Translation

5% damped spectral acceleration

Acceleration in g's

FIGURE 3.5-1
COMPARISON OF STRAIN-COMPATIBLE SHEAR MODULI WITH RANGE USED IN SSI ANALYSIS

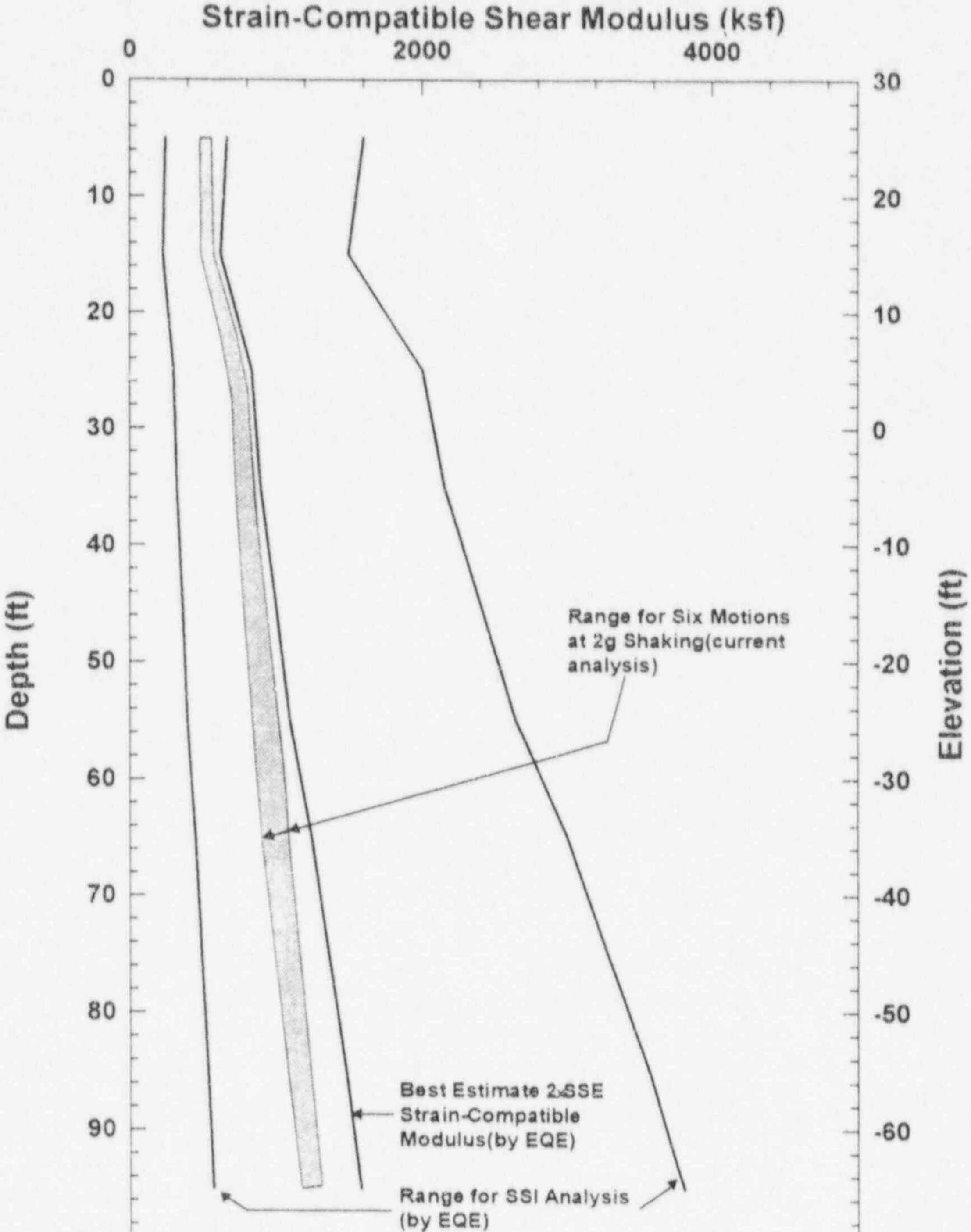
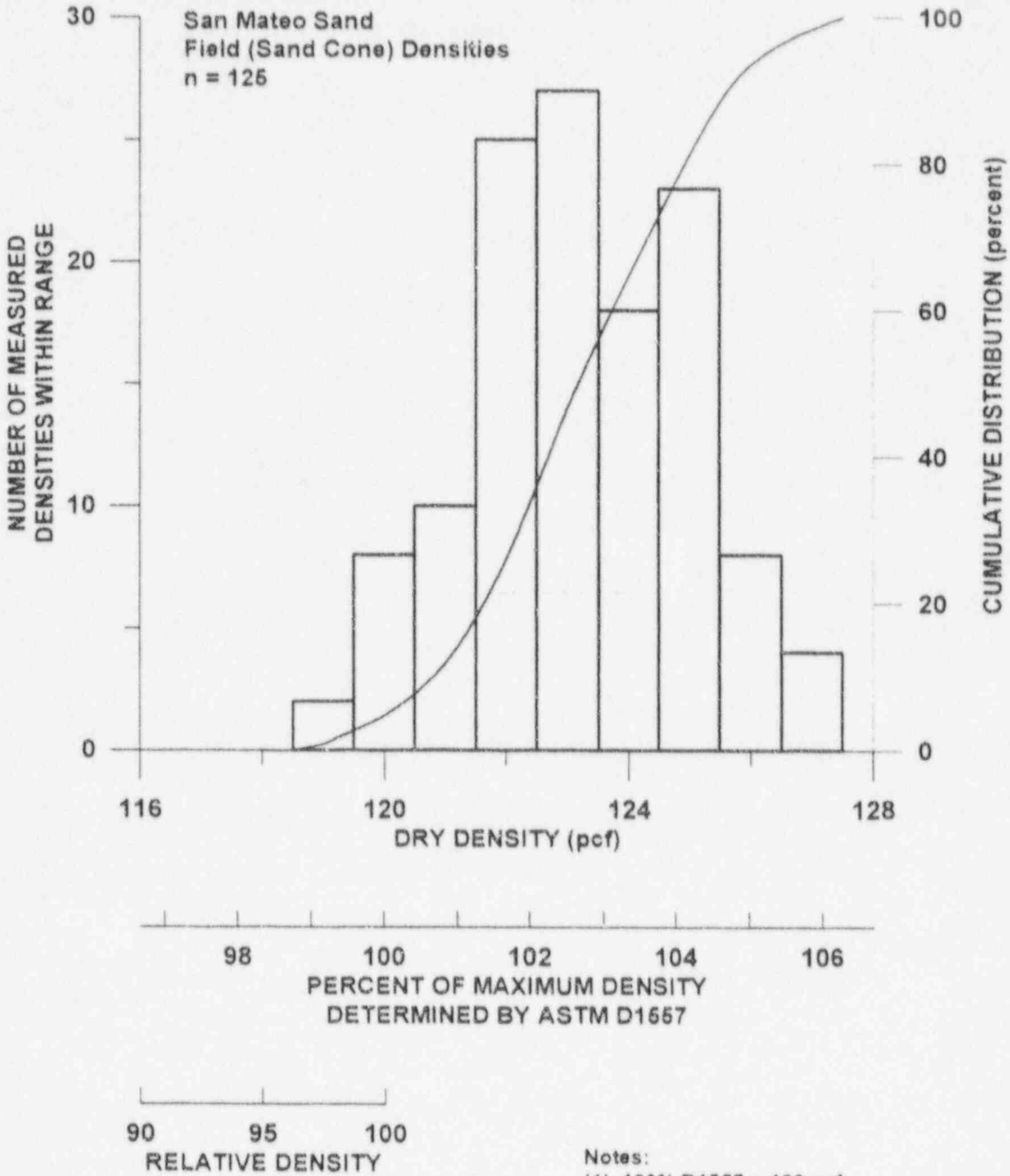
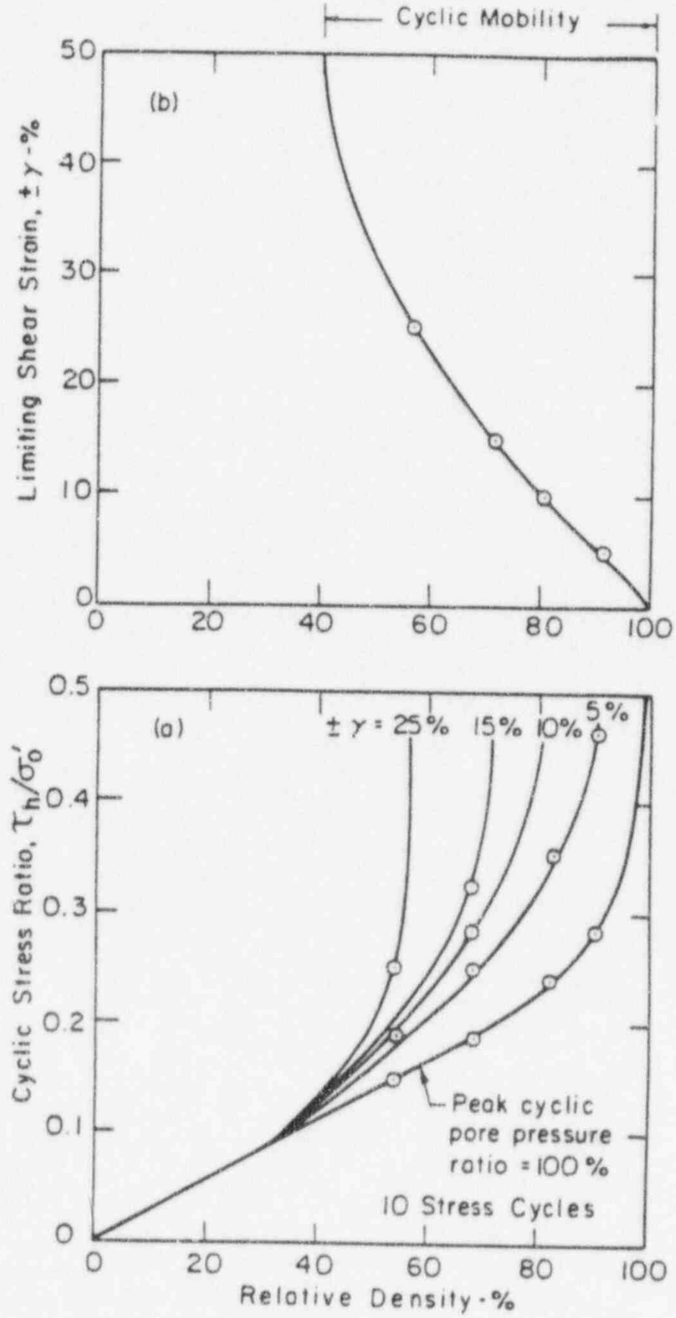


FIGURE 3.5-2
HISTOGRAM AND CUMULATIVE DISTRIBUTION OF IN-SITU DENSITIES IN SAN MATEO SAND



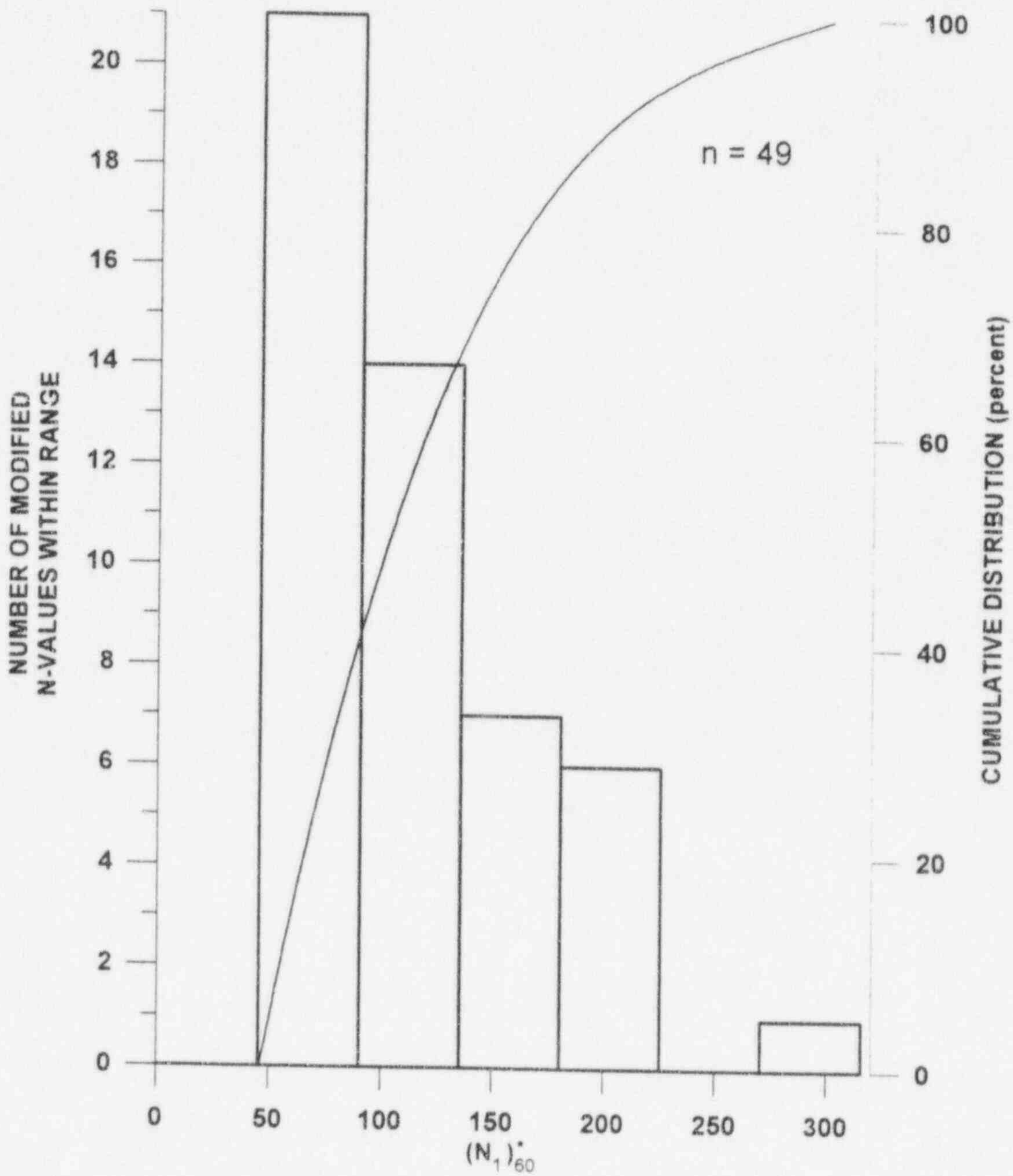
Notes:
(1) 100% D1557 = 120 pcf
(2) Relative Density based on
Min. Density = 89.1 pcf (avg. of two tests)
Max. Density = 120 pcf (max. lab density)

FIGURE 3.5-3
LIMITING SHEAR STRAINS FROM LARGE-SCALE LABORATORY SIMPLE SHEAR TESTS



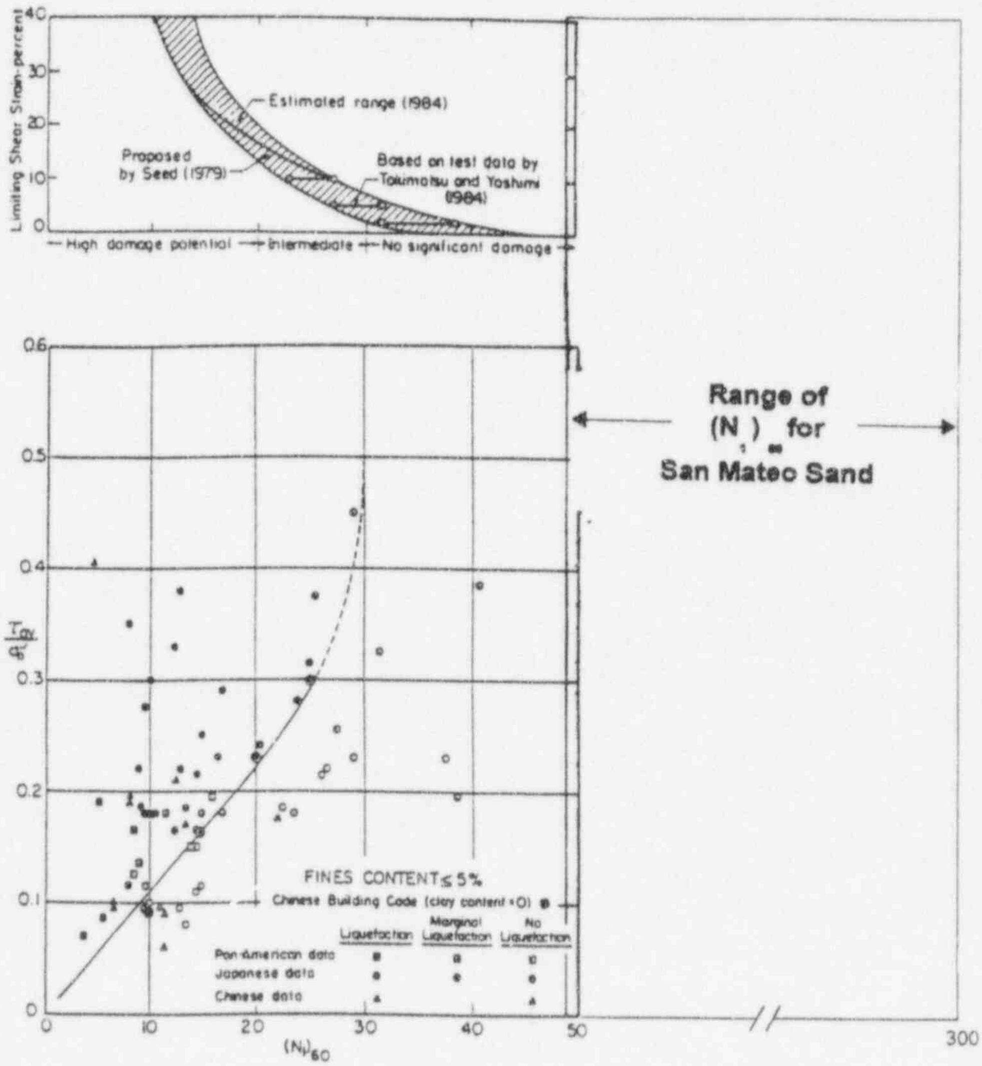
Source: De Alba, Seed, and Chan (1976)

FIGURE 3.5-4
HISTOGRAM AND CUMULATIVE DISTRIBUTION OF MODIFIED N-VALUES IN SAN MATEO SAND



* NOTE: Most blowcounts estimated by linear extrapolation to equivalent penetration of 12 inches.

FIGURE 3.5-5
EVALUATION OF LIQUEFACTION RESISTANCE USING SPT DATA



Source: Seed et al. (1984)
Magnitude 7.5 event

FIGURE 3.5-6
COMPARISON OF CYCLIC RESISTANCE VS. LIQUEFACTION THRESHOLD FOR SAN MATEO SAND

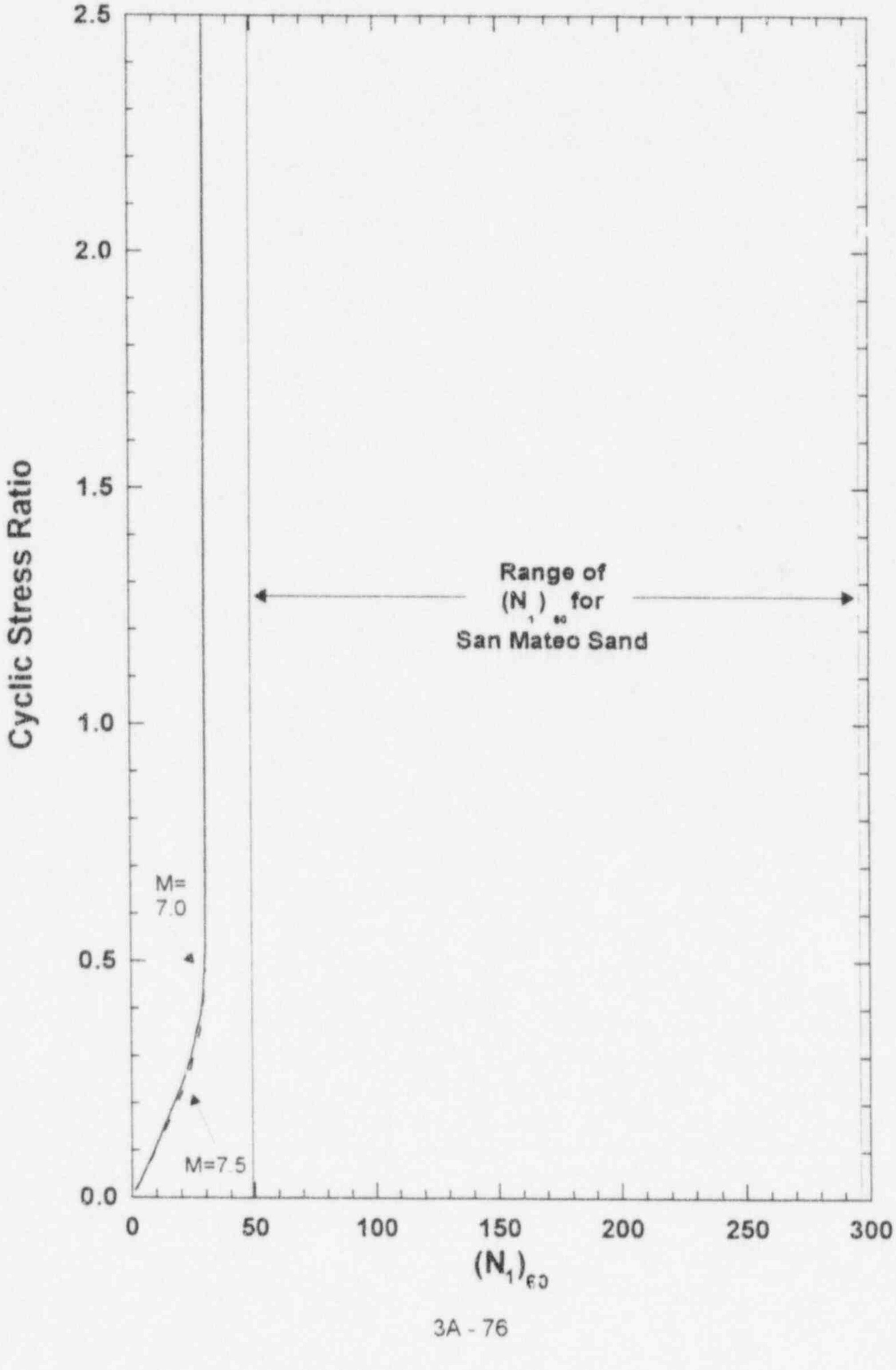
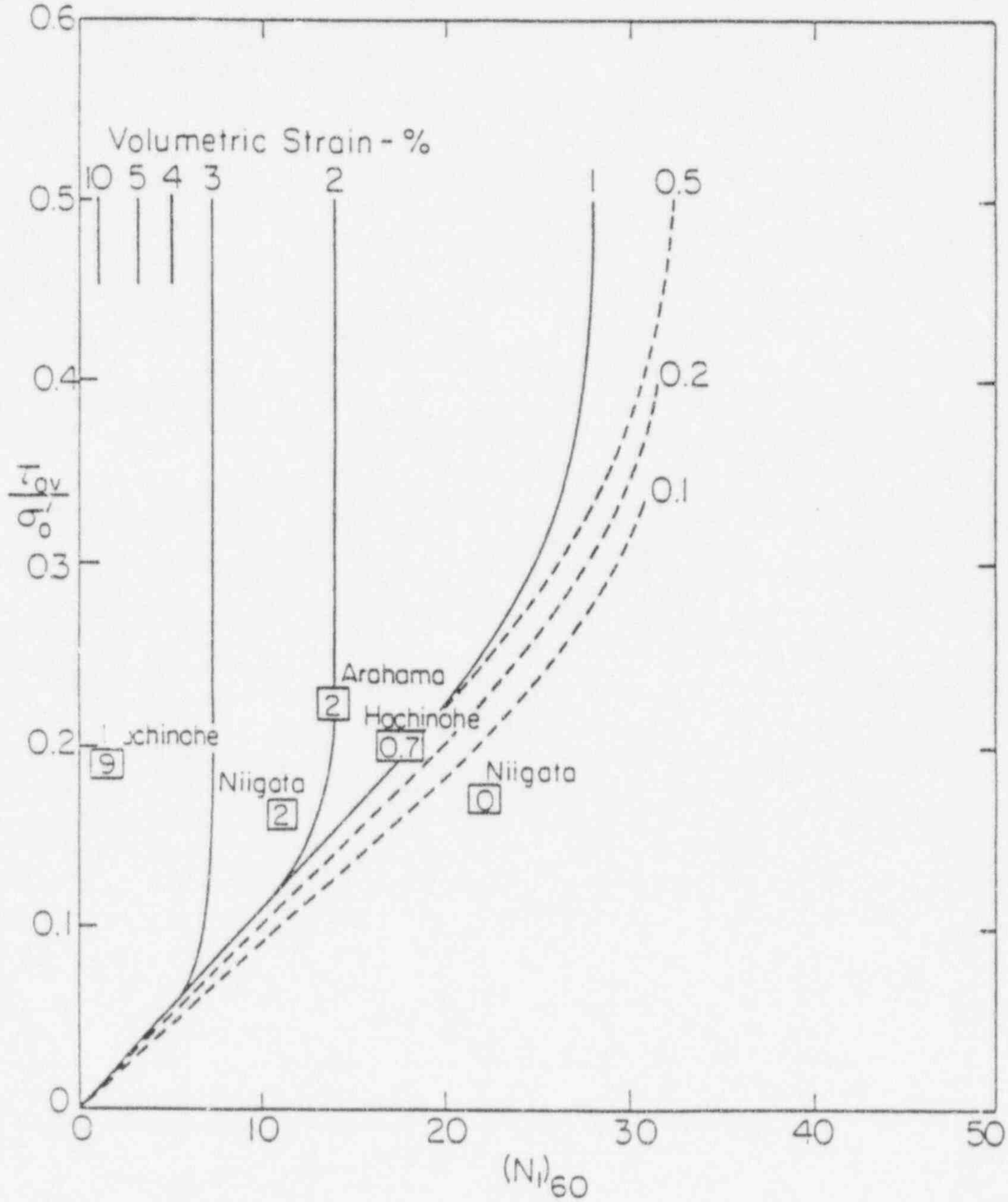


FIGURE 3.5-7
PROPOSED RELATIONSHIPS BETWEEN VOLUMETRIC STRAIN AND $(N_1)_{60}$



Source: Tokimatsu and Seed (1987)

FIGURE 3.5-8
COMPARISON OF RESULTS FOR SAN MATEO SAND WITH LIMITING SHEAR STRAIN TEST RESULTS

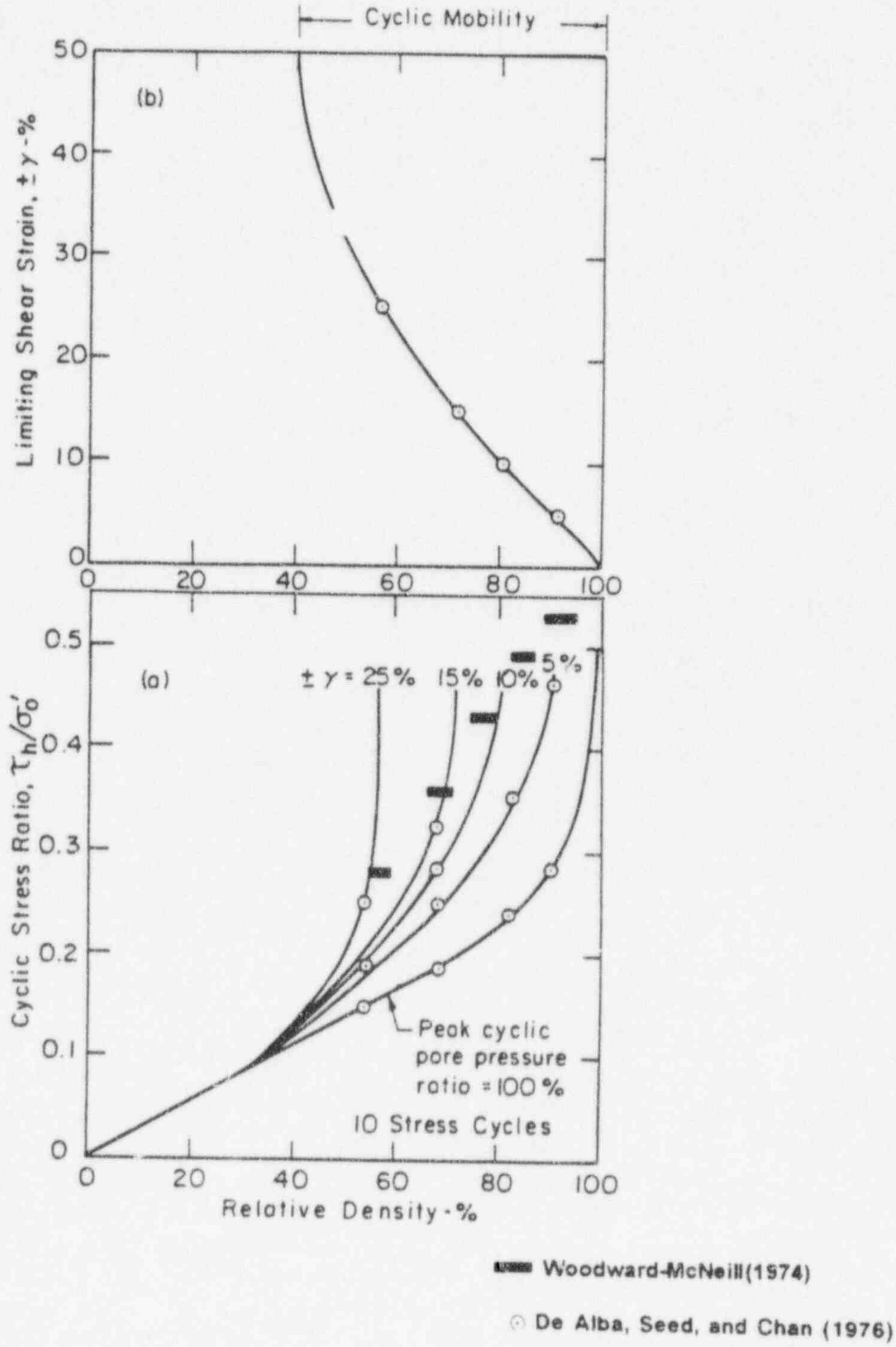


FIGURE 3.5-9
COMPARISON OF STRAIN-CONTROLLED CYCLIC STRENGTH VS. CALCULATED SHEAR STRAIN

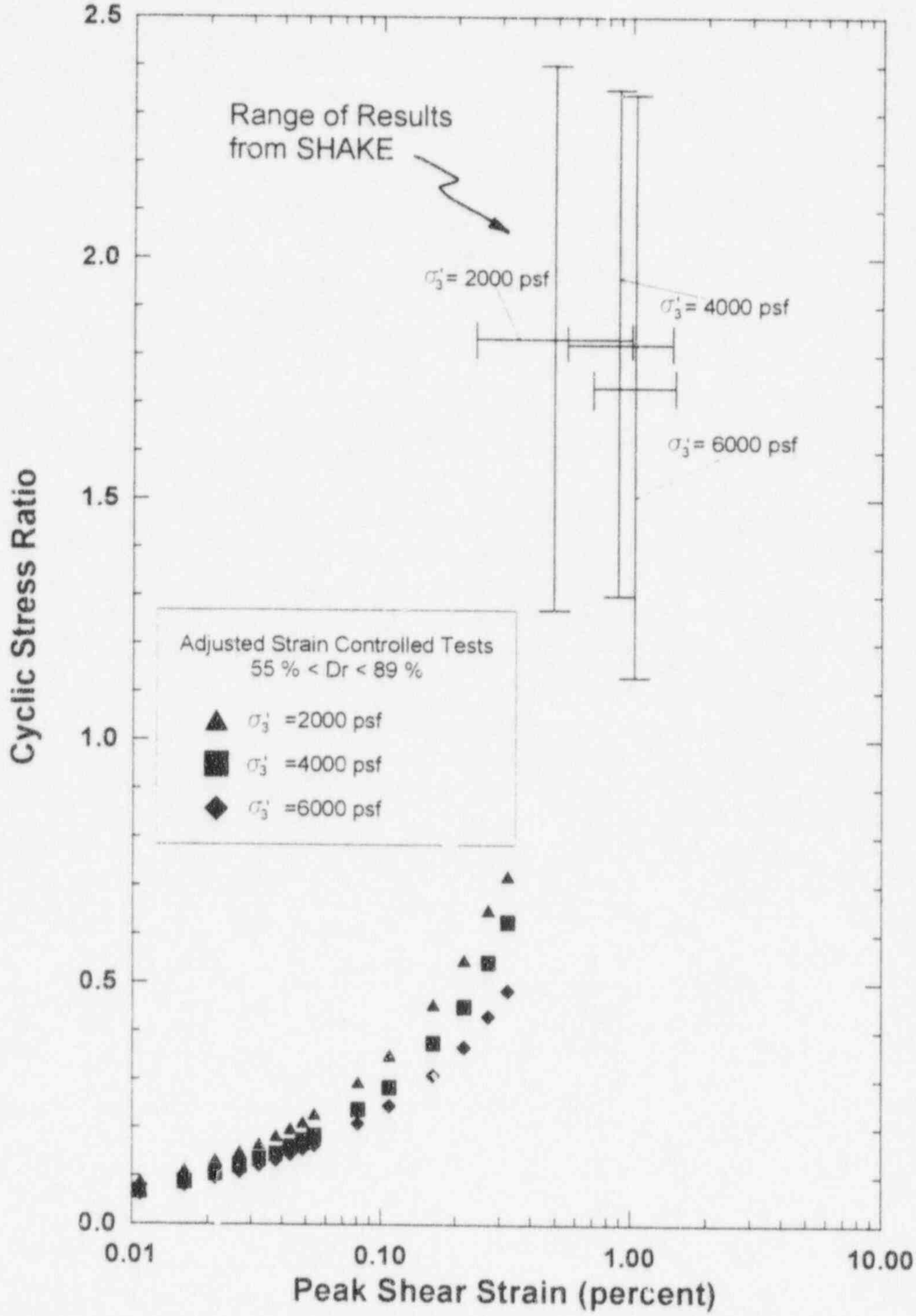


FIGURE 3.5-10
COMPARISON OF OTHER TEST RESULTS AND MEASURED RANGE OF PENETRATION RESISTANCE

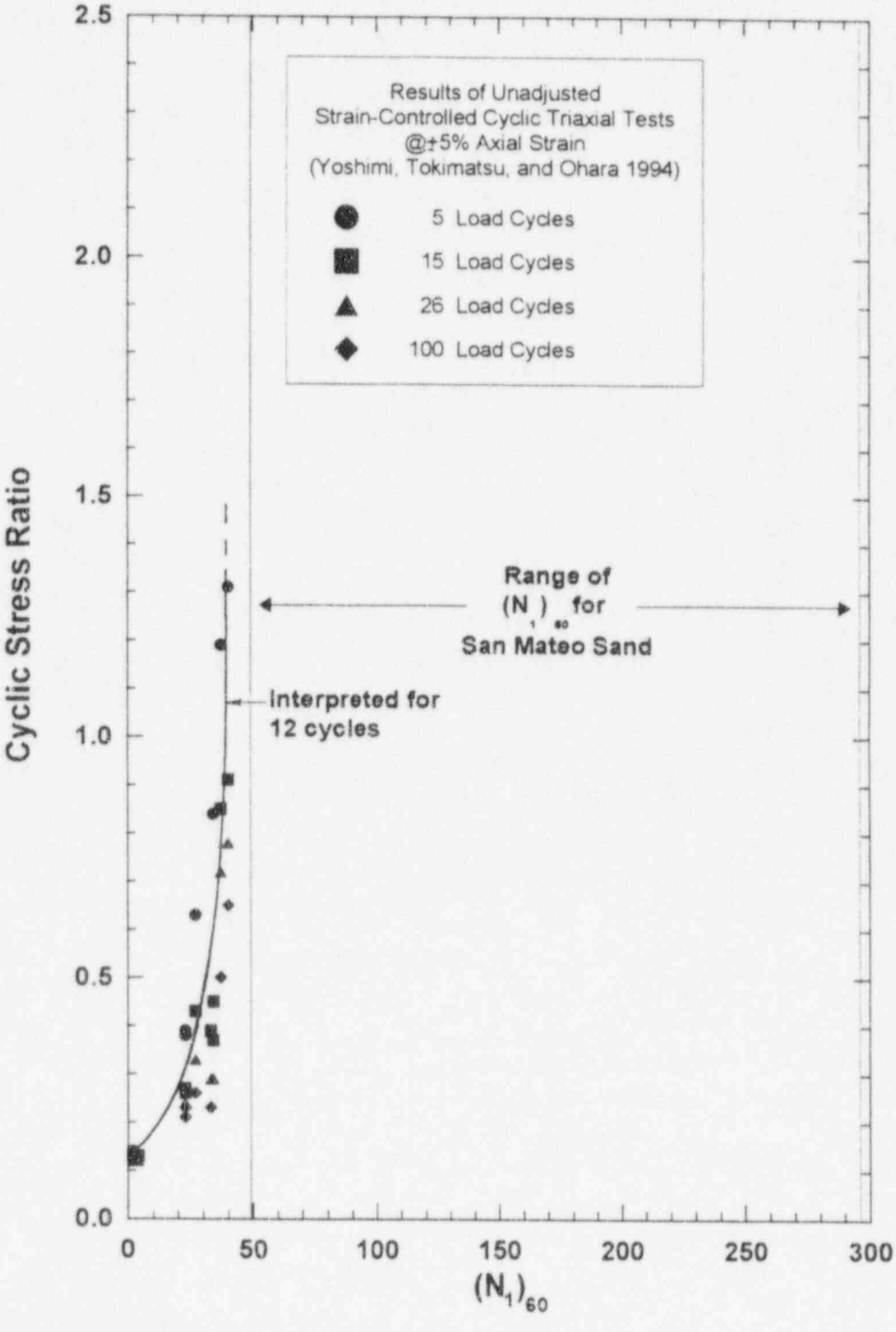
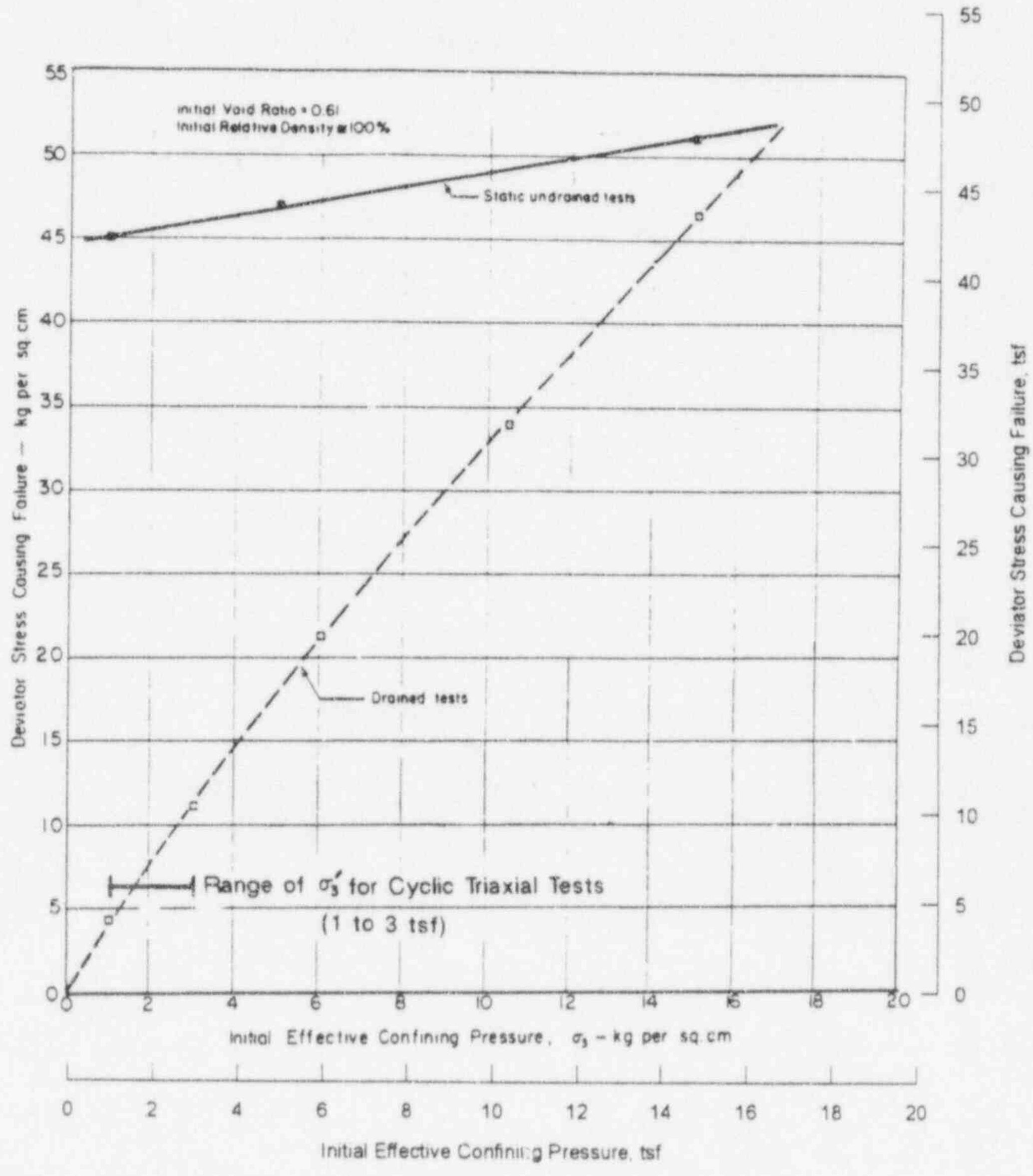
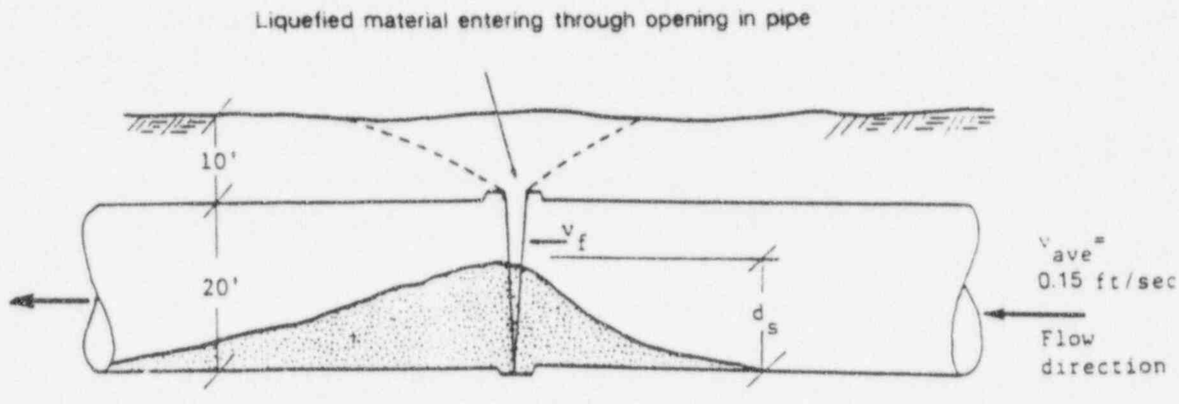


FIGURE 3.5-11
COMPARISON OF DRAINED AND UNDRAINED STRENGTHS
FOR D_r (INITIAL) = 100%

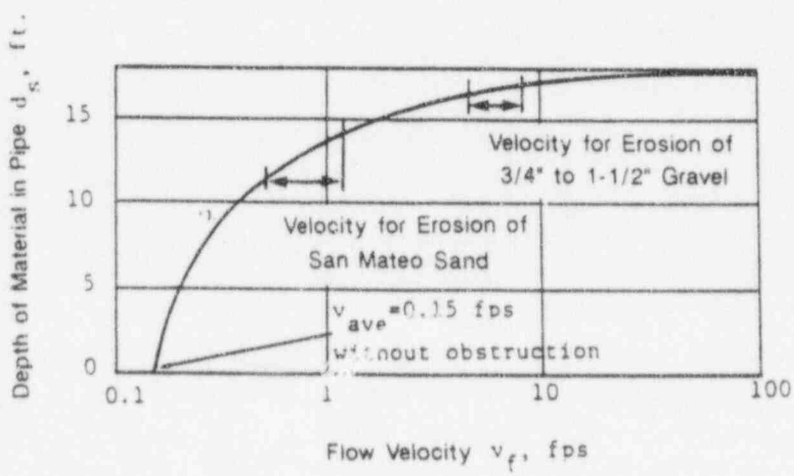


Source: Seed and Lee (1965)

FIGURE 3.5-12
SCHEMATIC DRAWING OF PIPE JOINT SEPARATION AND MAXIMUM DEPTHS OF SOIL ENTERING PIPE



Postulated Joint Separation



Increase in Velocity at the Obstruction

FIGURE 3.5-13
SLOPE STABILITY PROFILE AND SLIP SURFACE FOR 2:1 SWITCHYARD SLOPES

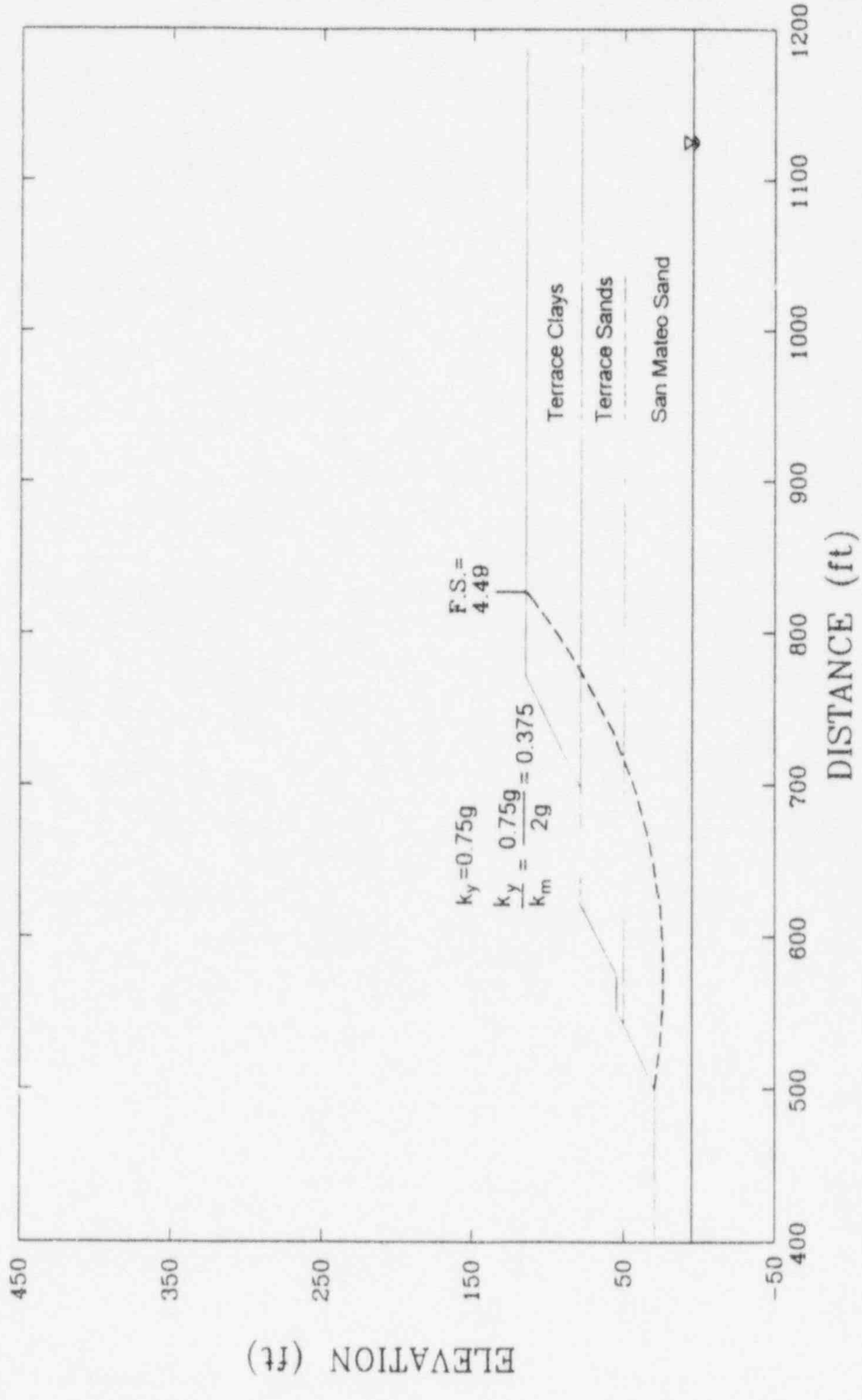


FIGURE 3.6-1
SEISMIC RISK QUANTIFICATION PROCESS

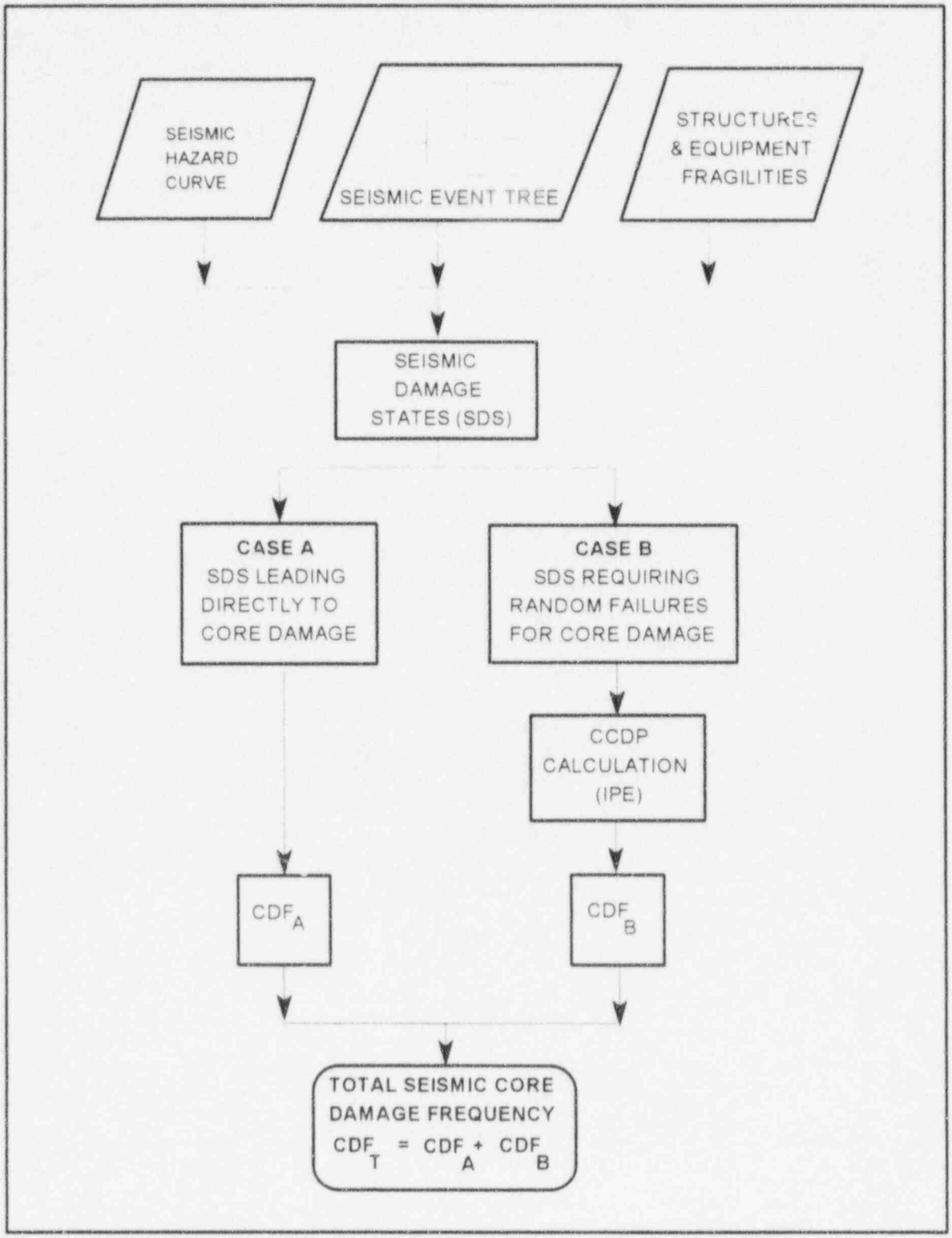
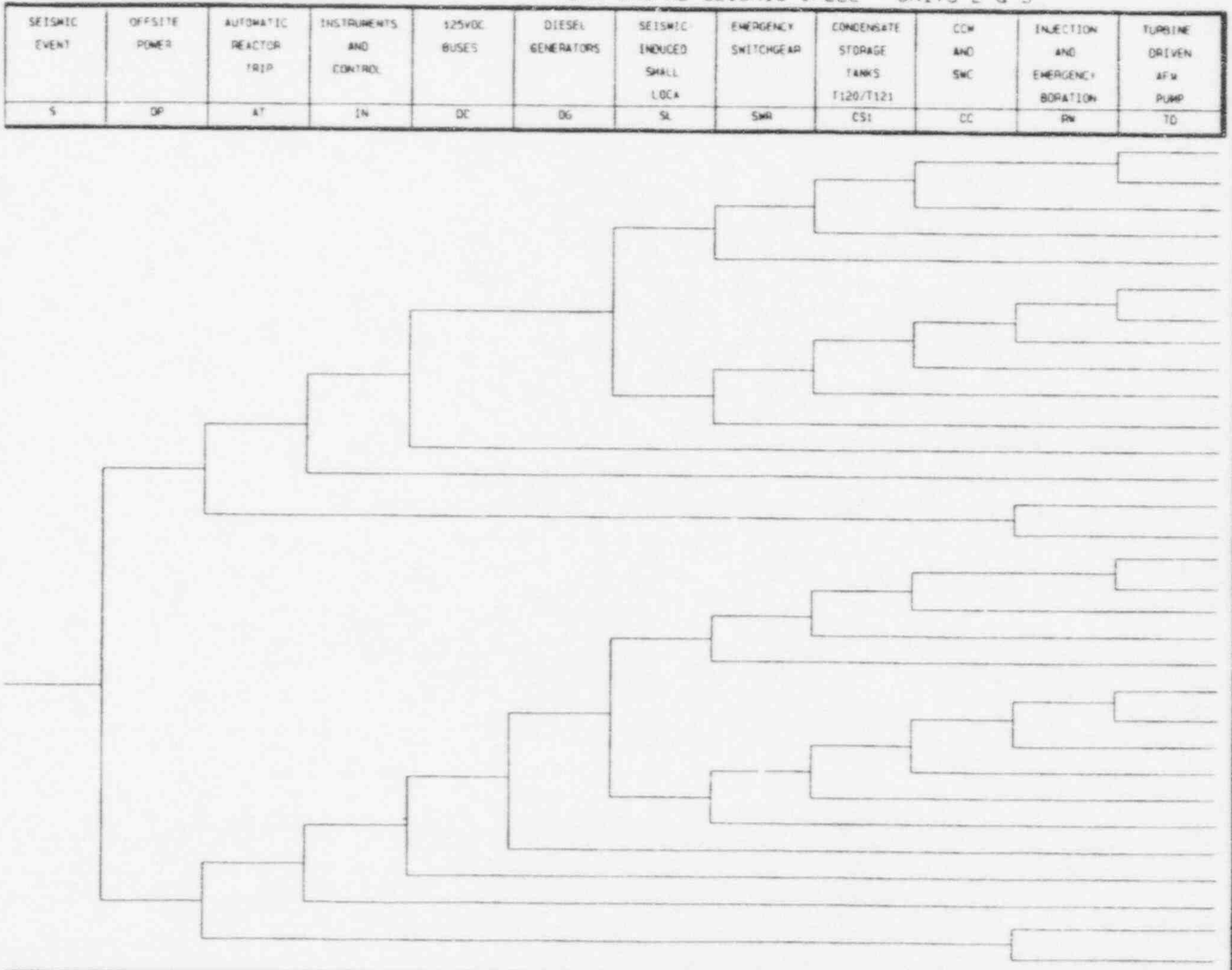


FIGURE 3.6-2

SONGS 2/3 SEISMIC EVENT TREE

SAN ONOFRE SEISMIC IPEEE - UNITS 2 & 3



NUMBER	SEQUENCE DESIGNATION	ACCIDENT CLASS	SEQUENCE FREQUENCY
1	S		
2	S-TD		
3	S-CC		
4	S-CS1		
5	S-SWR		
6	S-SL		
7	S-SL-TD		
8	S-SL-RW		
9	S-SL-CC		
10	S-SL-CS1		
11	S-SL-SWR		
12	S-DC		
13	S-IN		
14	S-AT		
15	S-AT-RW		
16	S-OP		
17	S-OP-TD		
18	S-OP-CC		
19	S-OP-CS1		
20	S-OP-SWR		
21	S-OP-SL		
22	S-OP-SL-TD		
23	S-OP-SL-RW		
24	S-OP-SL-CC		
25	S-OP-SL-CS1		
26	S-OP-SL-SWR		
27	S-OP-DG		
28	S-OP-DC		
29	S-OP-IN		
30	S-OP-AT		
31	S-OP-AT-RW		

REBECA (C) ERIN Engineering and Research, Inc. Version 3.10 SIE ET Name: SO2SIP 08-20-95 11:08:07

Total Frequency: 0.00E+00

FIGURE 3.6-3
SEISMIC EVENT MAGNITUDE VS. CDF

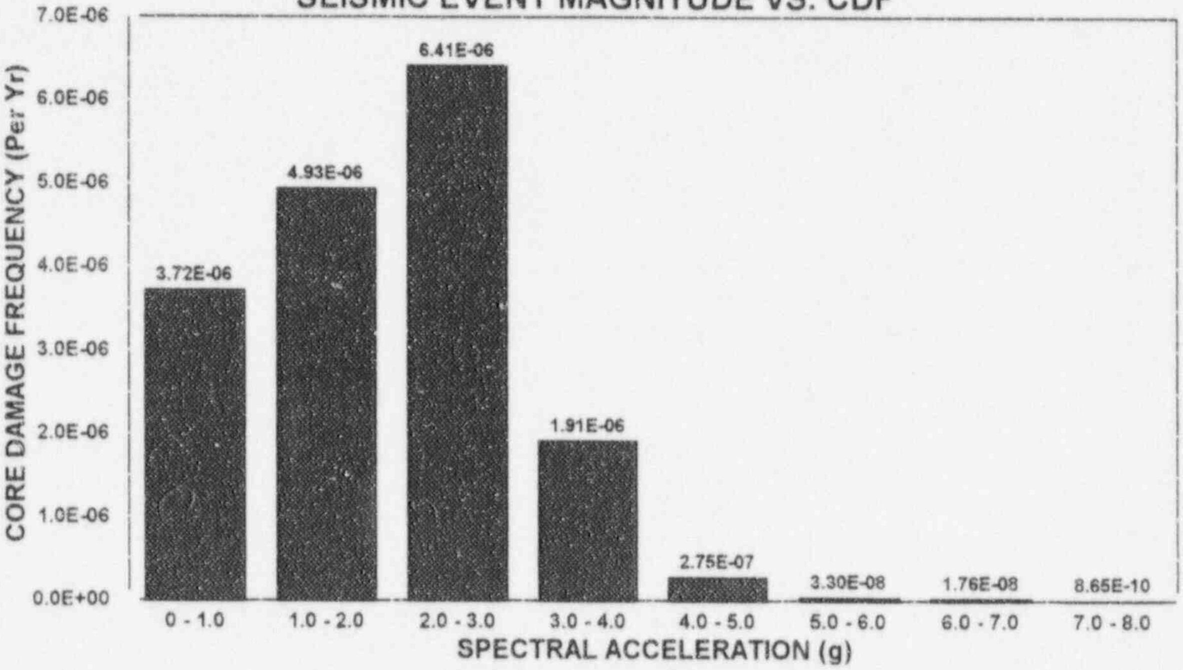
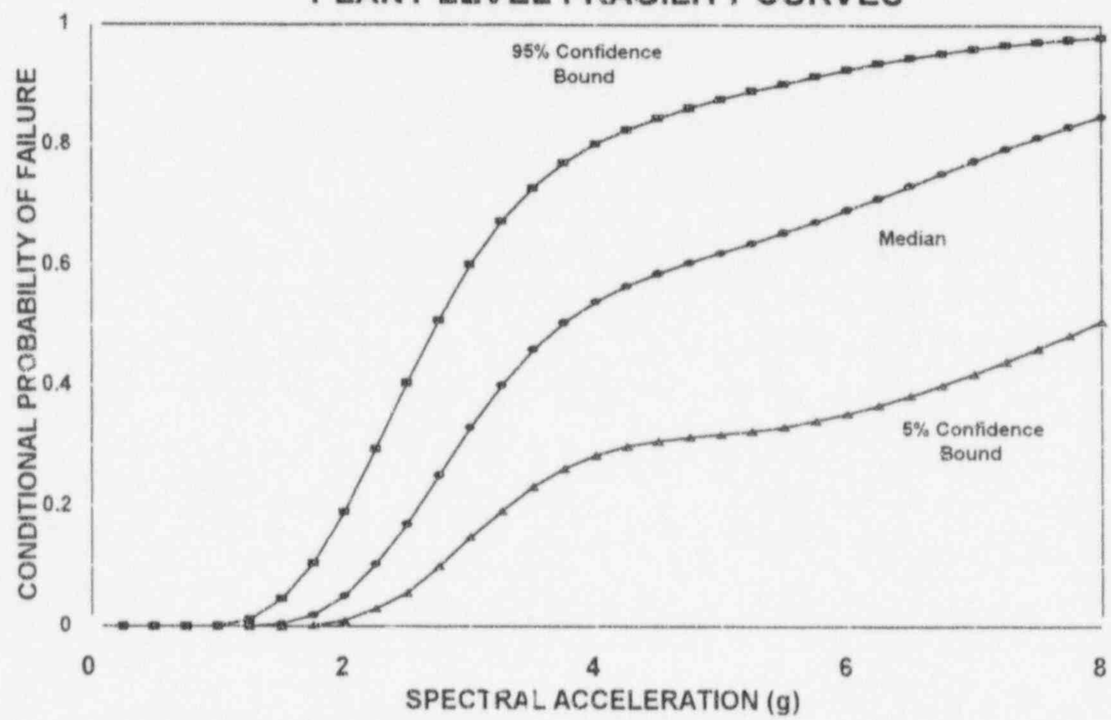


FIGURE 3.6-4
PLANT LEVEL FRAGILITY CURVES



4. INTERNAL FIRE ANALYSIS

This section provides a description of the methodology used to perform the internal fire analysis for SONGS 2/3, and a synopsis of the significant results for each phase of the analysis. The internal fire analysis fulfills the objectives of the IPEEE, and provides a systematic examination to identify any plant-specific vulnerabilities to severe accidents initiated by internal fire events. The organization of the section is:

Section 4-10 "Fire Appendix" provides floor layout schematics for the fire compartments. Since a combined EPRI Fire Vulnerability Evaluation (FIVE) and Fire Probabilistic Risk Assessment (PRA) was used to satisfy IPEEE requirements, the following table provides a cross-reference between the NUREG-1407 Standard Table of Contents and this submittal:

NUREG-1407	SONGS 2/3 Submittal
4.0 Methodology Selection	4.0
4.1 Fire Hazard Analysis	4.2
4.2 Review of Plant Information and Walkdown	4.1
4.3 Fire Growth and Propagation	4.3 (Simplified) 4.4 (Detailed)
4.4 Evaluation of Component Fragilities and Failure Modes	4.3 (Simplified) 4.4 (Detailed)
4.5 Fire Detection and Suppression	4.3 (Simplified) 4.4 (Detailed)
4.6 Analysis of Plant Systems, Sequences, and Plant Response	4.3 (Simplified) 4.4 (Detailed)
4.7 Analysis of Containment Performance	4.5
4.8 Treatment of Fire Risk Scoping Study Issues	4.6
4.9 USI A-45 and Other Safety Issues	4.7

The next section describes the methodology used for the internal fire analysis.

4.0 METHODOLOGY SELECTION

In conformance with NRC Generic Letter 88-20, Supplement 4, and NUREG-1407, the internal fire analysis used a combination of the two NRC-approved approaches, FIVE and fire PRA. The EPRI FIVE methods [4-1] were used for progressive screening of most fire compartments, and more detailed COMPBRN fire modeling [4-2] and PRA methods [4-3, 4-4] were used for the analysis of non-screened compartments. The overall process is depicted in Figure 4.0-1.

Phase I - Qualitative Screening Analysis

In Phase I, fire areas were screened out from further analysis if they did not contain Appendix R safe-shutdown components (or cables) and if a fire in that area would not cause a demand for safe-shutdown functions. Fire compartments, which are subsets of fire areas, were similarly screened out if they met the above criteria, and had no potential for fire spreading to other fire compartments.

Phase II - Simplified Fire Compartment Analysis

Fire compartments not screened out by the qualitative screening were evaluated using FIVE Phase II quantitative fire modeling. Fire risk was divided into three independent, multiplicative factors: the fire initiating event frequency, the unavailability of redundant or alternative equipment not affected by the fire, and the probability of sufficient combustible loading to damage critical compartment equipment before suppression. Each of these three factors was quantified in turn, and when the frequency of their combination evaluated to less than the FIVE criterion of 1E-6/yr, the compartment was screened out from further analysis.

Phase III - Detailed Fire Compartment Analysis

After the simplified quantitative fire compartment analysis was performed, the remaining compartments were evaluated using more sophisticated quantitative models consistent with NUREG/CR-2300, the PRA Procedures Guide [4-3]. A comprehensive set of fire scenarios was developed for each compartment, depending on the physical configuration of the compartment and the locations of combustibles, equipment, and cables. Computer models were developed for fire growth and suppression using COMPBRN IIIE, which accounts for software deficiencies identified in NUREG/CR-5088, the Fire Risk Scoping Study [4-5]. The internal events IPE models and data were modified to reflect the fire impacts, and to include potential recovery actions. Again, those compartments with total frequency less than the 1E-6/yr criterion were screened out from further evaluation. Note that the terminology of the "phased" approach adopted by the SONGS Fire IPEEE is slightly different than that stated in the FIVE methodology. The Phase III portion of this fire analysis involved the use of detailed fire modeling to reduce core damage frequency while the FIVE methodology identified Phase III as a walkdown and verification stage for closure of the fire risk assessment.

Containment Performance

Those compartments that did not screen out were evaluated to determine any unique containment performance issues, particularly with respect to the potential for containment bypass or early, large releases to the environment. The methodology used extended plant damage state event trees and fault trees, containment event trees, and source term grouping logic trees to fully quantify the dominant fire scenarios, and determine the impacts on containment performance and source terms.

Walkdowns, Fire Risk Scoping Study, and Other Issues

Walkdowns were performed to ensure that the fire models and information properly represent the as-built, as-operated plant. The checklists included in the FIVE methodology were used to address the Fire Risk Scoping Study issues. Other issues, including USI A-45 Decay Heat Removal, SECY-93-143, and seismic-fire interactions were specifically identified and discussed. More detail on the FIVE and PRA methods is given in References 4-1, 4-2, and 4-3, while details on implementation are provided in the following subsections.

FIGURE 4.0-1 SONGS 2/3 INTERNAL FIRE ANALYSIS

LEVEL 1 Impact on Plant Systems

PHASE I

Qualitative Fire Area Screening Analysis
- Area/compartment screen based on:
  * No safe shutdown components
  * No reactor trip or manual shutdown
- Fire compartment propagation analysis
- Phase I walkdown
- Quantitative bounding evaluation for FIVE rev. 1
- Containment fire area evaluation
- Unit 3 comparison

↓

PHASE II

Simplified Fire Compartment Analysis
- Screen out if < 1E-6/yr
  * Fire initiating frequency
  * Conditional core damage probability
  * Critical combustible loading probability using simplified fire models
- Phase II walkdowns

↓

PHASE III

Detailed Fire Compartment Analysis
- Screen out if < 1E-6/yr
  * COMPBRN fire models
  * Probabilistic scenario evaluations
  * Operator recovery actions

↓

LEVEL 2 Containment Performance Analysis

SYSTEMS ANALYSIS
- Representative fire scenarios
- Extended Level I event trees
- Plant damage state sequence assignment
- Containment bypass assessment

↓

SOURCE TERM ASSESSMENT
- Plant damage state logic trees
- Containment event trees
- Source term categories
- Cutsets of plant damages states and source term categories

↓

RESULTS

Dominant fire scenarios and contributors
Identification of potential vulnerabilities
Evaluation of potential modifications
Resolution of Fire Risk Scoping Study issues
Resolution of Decay Heat Removal, SECY 93-143, and other Generic Issues

FIREANAL.WPG

4.1 REVIEW OF PLANT INFORMATION AND WALKDOWN

This section provides a description of the primary sources of information used in the fire analysis, lists the status of pending fire protection modifications, and describes the walkdown procedures and findings.

4.1.1 PLANT INFORMATION SOURCES

An extensive list of plant documents and calculations were used to perform the fire analysis. These documents are listed in detail in the IPEEE Fire - Phase I and Phase II/III calculation documents [4-6, 4-7], and summarized below:

- Appendix R Documents (series 90035 and 90041), including:

  Power supplies
  Common enclosures
  Manual action feasibility
  Fire area boundary evaluation
  Fire protection engineering evaluations
  Technical Specification barrier review
  Fire area penetration seal evaluations
  Safe shutdown component evaluation
  Appendix R Database Management System (ARDMS)
  Detection / suppression / separation adequacy evaluation
  Alternative shutdown capability evaluation
  Auxiliary Gas System H_2 Storage and N_2 P&ID
  Cable routing
  Fire damper study
  Compliance assessment report
  SETROUTE Database
  Cable tray combustible loading
  Circuit analysis

- Updated Fire Hazards Analysis (UFHA)

- SONGS 2/3 Individual Plant Examination (IPE), Response to Generic Letter 88-20

- General SONGS 2/3 Information:

  Technical Specifications
  Elementary diagrams
  General arrangement drawings
  System descriptions
  Tray and conduit plans
  Piping material classification

- Procedures:

  SO23-13-2, Shutdown from Outside the Control Room
  SO23-13-21, Fire
  SO23-15-60A, Alarm Response Instructions
  SO123-XIII and -XV series procedures related to the fire protection program, training, and inspections

4.1.2 PROCEDURE MODIFICATIONS

The fire IPEEE was performed using the Cycle 7 plant configuration with the following upgrades included in the assessment for the detailed Phase III analysis.

1. For fire compartment 2-AC-30-20A (control room and cabinet area), implementation of an administrative change to Procedure SO23-13-2 (Shutdown From Outside The Control Room) would allow operators to utilize offsite power in the event that the reserve auxiliary transformers are not tripped by fire-induced damage to panel 2/3CR-63.

2. For fire compartments 2-DG-30-155 and 2-DG-30-158 (diesel generator rooms), implementation of an administrative change to Procedure SO23-13-21 (Fire) would allow operators to recover power to the 4 kV switchgear by disconnecting power to the diesel generator feeder breaker and reclosing the offsite power breaker on the switchgear.

3. Procedure enhancement for fire/seismic/internal events risk reduction. For fire compartments 2-AC-50-44, 2-AC-50-45, 2-AC-50-46, 2-AC-50-47 (distribution rooms), 2-AC-50-35 and 2-AC-50-40 (switchgear rooms), implementation of an administrative change to alarm response procedure SO23-15-60.A1 (Annunciator Panel 60A, Emergency HVAC) would allow operator to use air ducting and gas driven fans to prevent room heat-up.

In addition, in response to NRC Information Notice 89-52 [4-8], the plant completed procedure revisions in regards to manual actions for de-energizing fans to ensure fire damper closure and has revised the test procedures for dampers where manual actions may not be feasible. While HVAC and room ventilation operation is explicitly considered for the switchgear rooms, distribution rooms, and chiller rooms, operator actions to turn off the HVAC in order to ensure damper closure are not included in the fire models. Proceduralized operator actions to turn on emergency HVAC if normal HVAC is failed, or arrange alternative ventilation are included in the fire models. There are at least several hours between loss of ventilation and room heat-up equipment failures for the operators to restart ventilation, or to arrange alternative ventilation. Therefore, for the fire IPEEE, the procedures with respect to damper closure were implemented, and the HVAC recovery procedures were implemented and explicitly included in the fire model.

4.1.3 PLANT WALKDOWNS

SONGS 2/3 maintains an extensive set of documentation related to fire safety and plant configuration. While the fire IPEEE relied heavily on this base of documentation, plant walkdowns were performed to gather additional information needed for this assessment of potential severe accidents, and to ensure that there were no significant changes in plant configuration or as-operated condition. The following plant walkdowns were performed during the fire IPEEE process:

1. Phase I - Fire Compartment Interaction Walkdown (Propagation)

2. Phase II/III - Fire Modeling Walkdowns

3. Seismic-Fire Interaction Walkdown

Each of these are briefly discussed below, and the first two are documented in detail in the IPEEE Fire - Phase I and Phase II/III calculation documents [4-6 and 4-7]. The seismic-fire walkdown is documented in the seismic IPEEE section.

Phase I - Fire Compartment Interaction Walkdown

As part of the qualitative screening of fire compartments, it was necessary to demonstrate that a fire in one compartment would not spread to adjacent compartments. This "fire compartment interaction analysis" used the following systematic FIVE criteria (numbered 3, 4, 5, and 6) related to the amounts of in situ combustibles, fire barriers, and automatic fire detection or suppression to demonstrate that the compartment could be considered separate from adjacent compartments.

3) Boundaries that consist of a 1-hour rated fire barrier with a combustible loading in the exposing compartment <80,000 BTU/ft^2 on the basis of barrier effectiveness and combustible loading.

4) Boundaries where the exposing compartment has a very low combustible loading <20,000 BTU/ft^2 and automatic fire detection on the basis that manual suppression will prevent fire spread to the adjacent compartment.

5) Boundaries where both the exposing and exposed compartment have a very low combustible loading <20,000 BTU/ft^2 on the basis that a significant fire cannot develop in the area.

6) Boundaries where automatic fire suppression is installed over combustibles in the exposing compartment on the basis that this will prevent fire spread to the adjacent compartment.

Plant walkdowns were performed by the fire protection staff performing the Phase I screening for boundaries covered by the above criteria to ensure there were no concentrations of in situ combustibles near the boundary or in situ combustible pathways between compartments. Also, fire suppression provisions were assessed for those boundaries screened based on criterion 6.

Some fire compartment boundaries which were screened using criterion 5 were not walked down since the boundaries were either fire rated, constructed of hard concrete, or a previous fire boundary evaluation determined that there was no potential for fire spread between the subject fire compartments. Fire compartments boundaries using criterion 3 located in containment were not walked down due to accessibility constraints. Screening of these boundaries was acceptable based on low combustible loading in containment. Walkdowns were not performed for the comparable Unit 3 compartments as it was assumed that the results for these compartments would be the same as the results for the Unit 2 compartments. This assumption was based on the fact that the maximum permissible combustible loading values and suppression systems, as identified in the UFHA, are the same for both units for the compartments identified.

Based on these walkdowns, the electrical and cable tunnels were combined into one fire compartment, and the main turbine building and SWC pump room were combined to form a new fire compartment to ensure that there would be no fire propagation between compartments.

Phase II/III - Fire Modeling Walkdowns

Several walkdowns were performed by the fire protection staff performing the Phase II and III analyses and fire modeling tasks. Their primary objectives were:

- Gather or verify spatial information, such as separation distances between fire sources and targets, and tray heights, in order to perform the simplified fire modeling for Phase II, and the detailed COMPBRN fire modeling for Phase III.

- Identify and verify potential fixed and transient fire sources, and confirm conclusions such as small fire sources incapable of damaging safe shutdown equipment or cables.

- Determine whether cabinets were ventilated or not.

The information from these walkdowns was incorporated into the fire analysis and documented in the calculation files. No significant deviations from the plant design and configuration information were noted during the walkdowns.

Seismic-Fire Interaction Walkdown

A combined seismic-fire and seismic-flood interaction walkdown was performed by a team composed of fire protection staff performing the fire IPEEE, seismic capability staff performing the fragility portion of the seismic IPEEE, and PRA staff performing the probabilistic and systems portions of the fire and seismic IPEEE. This walkdown is described in more detail in the seismic IPEEE documentation. No seismic-fire or seismic-flood interactions were identified that would require additional analysis.

4.2 FIRE HAZARD ANALYSIS (FIVE PHASE I)

This section provides a description of the methods, assumptions, and results of the qualitative FIVE Phase I screening analysis for SONGS 2/3. Also included is the evaluation of potential fires in containment (Section 4.2.2), and a comparison of Unit 2 and Unit 3 similarities and differences with respect to the fire IPEEE (Section 4.2.3).

4.2.1 PHASE I QUALITATIVE SCREENING ANALYSIS

The first phase in the fire analysis was to perform a qualitative screening of major fire areas and fire compartments at the plant. The FIVE method, (Revision 1), allows a fire area to be screened out of the analysis if:

a. There are no Appendix R safe-shutdown components in the fire area.

AND

b. Following a fire event in a fire area, there is no demand for safe-shutdown functions because the plant can maintain normal plant operations.

Note that FIVE Revision 0 allowed screening if either of these criteria were met. The original Phase I analysis was performed before Revision 1 was issued. A conservative alternative to completely reperforming a detailed Phase I evaluation was chosen. The compartments screened per Revision 0 were subjected to a high level re-evaluation per Revision 1 criteria. Only those compartments which could easily be shown to meet Revision 1 criteria remained screened. For expediency, any compartment requiring extensive research to demonstrate meeting the screening criteria was left unscreened and was subjected to a quantitative analysis using Phase II methods. This conservative approach is described in Section 4.2.1.9.

As with the Appendix R analysis, FIVE differentiates between fire areas (whose boundaries have been analyzed to ensure strict non-propagation of fires beyond the fire area), and fire compartments, termed zones in the Appendix R analysis. Fire compartments are generally subdivisions of a larger fire area, and are based on the presence of non-combustible barriers that would substantially confine heat and products of combustion, but may not meet the Appendix R criteria for boundaries. Appendix R related materials were used extensively to provide information for this definition of fire compartments, and for the screening process. Walkdowns were performed to verify the information in the plant documentation.

The FIVE methodology for Phase I was used as a guideline, although some changes were made in order to fully utilize the extensive SONGS 2/3 databases, and to reflect plant-specific features. As mentioned above, the screening process was performed in two stages: screening using FIVE Revision 0, and screening verification using bounding quantitative analysis to meet FIVE Revision 1 changes. The methodology was divided into the following tasks:

1. Identify fire areas and fire compartments.
2. Identify safe shutdown systems.
3. Identify systems requiring shutdown.
4. Perform cable routing for non-Appendix R components.
5. Screen fire areas without the demand for shutdown.
6. Identify fire compartments with the demand for shutdown.
7. Perform the fire compartment interaction analyses for compartments in Task 6.
8. Identify Phase II fire compartments.
9. Evaluate FIVE methodology Revision 1 changes

Each of these tasks is discussed briefly, with particular attention to assumptions or exceptions to the FIVE methodology. Unless specified otherwise, this methodology applies to Unit 2 and Common areas and components.

4.2.1.1 Identify Plant Fire Areas and Plant Fire Compartments

The purpose of this step was to divide the plant areas into sections for separate analyses. The identification of plant fire areas and fire compartments was performed consistent with the FIVE definitions for fire areas and fire compartments (Refer to FIVE Sections 7.1 and 7.3). Section 4.10 provides floor layout drawings showing these fire compartments. Fire areas were defined as being completely surrounded by fire rated barriers which are surveyed per technical specifications. There were a few exceptions to the FIVE criteria, all of which had specific boundary analyses performed for Appendix R:

a. Boundaries which were at grade (such as the outside yard) or which were external walls and have been shown to provide adequate protection to prevent fire propagation are considered fire area boundaries.

b. The boundary between the roof of the Safety Equipment Building and the Turbine Building was considered a fire area boundary based on existing barrier evaluations.

c. The boundary between the Unit 2 Turbine Building and the Unit 3 Turbine Building was considered a fire area boundary based on an existing separation evaluation.

d. Barriers which were not fire rated but had surveillance requirements and Appendix R boundary evaluations were considered to be acceptable fire area boundaries. These boundaries are typically constructed of heavy concrete.

e. The boundary between fire area 2-AR-44 and fire area 2-FH-47, separating fire compartment 2-FH-30-174A from fire compartments 2-FH-17-123, 2-FH-30-126, and 2-FH-45-130, does not meet the FIVE criteria for a fire area boundary. However, this boundary is considered a fire area boundary based on a 3 hour rated barrier and no combustibles, safe shutdown equipment susceptible to fire damage, or systems requiring shutdown in fire compartment 2-FH-30-174A, and no systems requiring shutdown in fire compartments 2-FH-17-123 and 2-FH-45-130.

f. The boundary between fire area 2-SE-TK-53 and fire area 2-AC-39, separating fire compartment 2-SE-70-172 from fire compartment 2-AC-70-64, does not meet the FIVE criteria for a fire area boundary since it is not surveyed by Technical Specifications. However, this boundary is considered a fire area boundary based on a 3 hour rated barrier separating the two fire areas, no in situ combustible loading, safe shutdown equipment, or systems requiring shutdown in 2-SE-70-172, and no systems requiring shutdown in 2-AC-70-64.

g. The containment fire compartment boundaries for 2-CO-15-1A, 2-C0-15-1B, 2-CO-15-1C, and 2-CO-15-1D, do not meet the FIVE criteria for a fire compartment boundary due to wall openings. However, these boundaries are considered adequate to prevent the spread of fire despite these wall openings, since they are constructed of 48 inch thick concrete. In addition, fire protection features for redundant equipment in fire compartments 3-CO-15-1C and 3-CO-15-1D were determined to provide a level of protection equivalent to the requirements of Appendix R, III.G.2.

4.2.1.2 Identify Safe Shutdown Systems

Safe shutdown systems are those plant systems required to achieve and maintain safe shutdown in the event of a fire. Safe shutdown systems are covered in the SONGS 10CFR50 Appendix R Safe Shutdown Analysis. In accordance with the FIVE Methodology, fire areas and compartments containing these systems are potentially more critical to plant safety during fire initiated events. The purpose of this task was to identify safe shutdown systems and determine their locations throughout the plant.

Document 90035AH, the SONGS 2/3 Safe Shutdown Component Evaluation of the UFHA [4-9] was reviewed in order to determine all plant systems credited for safe shutdown. These systems are listed below.

SSD System	Description
120V	120V AC power
125V	125V DC power
4160V	4160V AC power
480V	480V AC power
AFW	Auxiliary Feedwater System
CCW	Component Cooling Water System
CVCS	Chemical and Volume Control System
DG	Diesel Generator System

ECW	Emergency Chilled Water System
EP	Electrical Panels
ESF	Engineered Safety Features Actuation System
HVAC	Essential Heating, Ventilation and Air Conditioning System
MFW	Main Feedwater System
MSS	Main Steam System
RCS	Reactor Coolant System
RPS	Reactor Protection System
SDC	Shutdown Cooling System
SWC	Saltwater Cooling System

Safe shutdown components which comprise each system are also listed in the Safe Shutdown Component Evaluation. A cross-referenced list of safe shutdown components for each fire compartment was obtained using the SONGS 2/3 ARDMS database. These lists contain safe shutdown components which either (1) are located in the fire compartment, or (2) have associated cables located in the compartment. These lists are contained in Tier 2 documents and were used to identify safe shutdown systems and components for each fire area and fire compartment.

4.2.1.3 Identify Systems Requiring Shutdown

Plant systems whose fire-induced failure could cause the demand for safe shutdown were defined as systems which meet one of the following criteria:

- The system contains components which could be rendered inoperable due to a fire; and an inoperable status for the components requires a near term shutdown per the SONGS Technical Specifications.

- The system contains components whose fire-induced loss could initiate a reactor trip.

Per the FIVE Methodology, fire areas and compartments containing these systems are potentially more critical to plant safety during fire initiated events. As a result, fire areas and fire compartments which contain these systems cannot be screened out in FIVE Phase I screening. The purpose of this task was to identify plant systems which require shutdown, and determine their locations throughout the plant.

Technical Specification Near Term Shutdown Systems

SONGS 2/3 Technical Specifications were reviewed in order to identify components which could be rendered inoperable due to a fire, resulting in the need to perform a near term shutdown. Near term shutdown requirements are defined based on the guidelines presented in the FIVE Methodology as follows:

- Component that are inoperable as defined by surveillance requirements.
- Component whose action statement requires near term shutdown independent of restoration.
- Component whose action statement requires a restoration time of 8 hours or less.
- Component whose action statement requires unit to be placed in hot standby within 8 hours.
- Loss of components or instrumentation used to perform Technical Specification surveillances were not considered since alternate means would be utilized, if necessary.

The results of this review of near term shutdown requirements are contained in Tier 2 documentation.

For example, Technical Specification 3.8.2.1 requires that if one battery bank is unavailable, and is not restored in 2 hours, then the plant must be in hot standby in 6 hours. Each of the Technical Specifications was reviewed to determine which component losses would require near-term shutdown.

Reactor Trip Initiators

SONGS 2/3 Technical Specification for Reactor Protective Instrumentation Section 3/4.3.1 was reviewed in order to identify components whose fire-induced loss could initiate a reactor trip. These reactor trip initiators are contained in the Tier 2 documentation. Typical reactor trip initiators are SIAS, high or low steam generator water level, and turbine trip.

Note that although this task addresses the primary mechanisms which may result in a reactor trip, it may not be comprehensive in its identification of all mechanisms whose failure may eventually lead to a reactor trip. For example, damage to peripheral balance of plant components not included above may eventually cause a demand for reactor trip. However, based on the evaluation performed in Section 4.2.1.9, "Evaluate FIVE Methodology Revision 1 Changes," it was determined that any areas containing unanalyzed trip initiators have been adequately bounded. The evaluation performed in Section 4.2.1.9 revisited all fire compartments which screened out in Phase I but contained safe shutdown equipment. Using appropriate fire frequency and conditional core damage probability values from the Phase II evaluations, the methodology of 4.2.1.9 assigned a core damage probability value to those areas *conservatively assuming* that a reactor trip had occurred. Therefore, Section 4.2.1.9 addressed the worst case scenarios for unanalyzed trip initiators; and, as shown in Section 4.2.1.9, in all cases the areas were justifiably screened out in the initial Phase I.

4.2.1.4 Perform Cable Routing For Non-Appendix R Components

In accordance with the FIVE methodology, some systems requiring shutdown, or used to achieve and maintain shutdown, contained components which were not listed in the SONGS 2/3 Appendix R Safe Shutdown Component Evaluation. Cable and component location information in these cases was not available in the Appendix R III.G/III.L Compliance Assessment or the ARDMS database. The purpose of this task was to identify the cables associated with each of these non-Appendix R components, and determine their locations throughout the plant.

A list of Appendix R components in the Safe Shutdown Component Evaluation was compared against the list of components requiring near term shutdown or causing a reactor trip. The fire-induced unavailability of any of the following components require a near term Technical Specification shutdown but were not analyzed in the Appendix R safe shutdown analysis:

- Containment Emergency Sump isolation valves: HV-9302, HV-9303, HV-9304, and HV-9305, and

- ECCS mini-flow valves: HV-9306 and HV-9347.

In addition, failure of the following components could initiate a reactor trip but were not analyzed in the Appendix R safe shutdown analysis:

- Control Element Drive Mechanisms (CEDMs)

- Flux monitors JI-0002A,B,C,D and JI-OOOIA,B,C,D

- Core Protection Calculator (CPC) Inputs

- Turbine trip cables
- Pressure differential transmitters (PDT) for RCS flow: PDT-0978-1,2,3,4 and PDT-0979-1,2,3,4
- Triaxial accelerometers XS-8031-1,2,3,4

Cable identification and routing were performed for the following non-Appendix R components:

- Containment Emergency Sump (CES) isolation valves: HV-9302, HV-9303, HV-9304, and HV-9305
- ECCS mini-flow valves HV-9306, HV-9347
- Control Element Drive Mechanisms (CEDMs): CEDM12, CEDM37, CEDM47, CEDM57 and CEDM63
- Turbine trip cables
- Pressure differential transmitters for RCS flow: PDT-0978-1,2,3,4 and PDT-0979-1,2,3,4

Elementary diagrams for components listed above were reviewed to determine the applicable cables whose fire-induced failure could result in an undesirable condition for the components (i.e., results in a reactor trip or component Technical Specification inoperability). Cable identification was performed consistent with the requirements of Appendix R using the following conventions:

(1) Cable faults considered were: open circuit, wire to wire, short to ground, hot short.

(2) Three phase hot shorts and DC double hot shorts of the proper polarity were not considered.

(3) Cable fault consequences considered were loss of control, loss of power, and spurious operation.

Raceway routing of each cable was determined using SETROUTE. Each raceway was then located on the applicable tray and conduit plans. The UFHA Technical Specification Barrier Drawings were then reviewed to determine the fire compartment location of each raceway.

Cable identification and raceway routing was performed for a sample of 5 CEDMs. Based on these results, it was judged that CEDM routing was consistent for all CEDMs from the Containment Building to the CEDM cabinets.

Cable identification was also performed for offsite power to Unit 2 and Unit 3, POS4 and POS17, respectively. Based on a review of Document 90035BC, Units 2 and 3 System/Component/Cable ARDMS Report, the list of cables associated with POS4 and POS17 was found to be too conservative. That is, many of the cables identified in the Appendix R analysis would not actually cause a loss of offsite power if damaged. Circuit analysis was performed to determine the actual list of cables which impact offsite power. Using the methodology of this section, a revised list of cables and associated cable routing which impact POS4 and POS17 was identified.

Cable identification and routing was not performed for the following components:

- Flux monitors JI-0002A,B,C,D and JI-OOO1A,B,C,D
- CPC Inputs
- Triaxial accelerometers: XS-8031-1,2,3,4

Cabling associated with these components was assumed to follow routes similar to the routes of other components located in containment such as the containment emergency sump isolation valves and the pressure differential transmitters covered above. This assumption is also acceptable since none of the fire compartments between Containment and the control room screened out based on other component losses.

The results of the additional cable identification and routing for non-Appendix R components are documented in the Tier 2 documentation.

4.2.1.5 Screen Fire Areas Without The Demand For Shutdown

The initial FIVE methodology, Rev. 0, allowed fire areas to be screened from further consideration in Phase I if the fire area met either of the following criteria:

(1) The fire area contains no safe shutdown equipment.

(2) The fire area contains no components which will result in a reactor trip or near term shutdown requirement.

The SONGS 2/3 fire analysis initially used criterion (2) as the bounding criteria to ensure that potentially significant fire areas were not screened out. A total of 16 fire areas screened out based on meeting criterion (2), using the information developed in the previous tasks. A list of the fire areas screened out in Phase I is contained in Table 4.2-1.

TABLE 4.2-1
FIRE AREAS SCREENED OUT IN PHASE I

Fire Area	Fire Area Description
2-AC-70-67	Auxiliary Control Building - HVAC Duct Shaft
2-AC-85-180	Emergency Chilled Water Pipe Tunnel
2-AC-9-8	Auxiliary Control Building - Lighting Switchgear Room
2-AC-9-10	Auxiliary Control Building - Normal Chiller Room
2-AC-9-12	Auxiliary Control Building - HVAC Room
2-AC-9-13	Auxiliary Control Building - Lighting Switchgear Room
2-AC-16	Auxiliary Control Building - Corridor/Stair
2-AC-30-23	Auxiliary Control Building - Fan Room
2-AC-30-24	Auxiliary Control Building - Staircase
2-AC-50-41	Auxiliary Control Building - Distribution Room
2-AC-85-72	Auxiliary Control Building - Fan Room
2-AC-39	Auxiliary Control Building - Heath Physics/Access Control Areas
2-SE-(-5)-135D	Train A CCW Pump Room
2-DG-30-157	Diesel Generator Building - Staircase
2-AC-(-5)-169	Emergency Chilled Water Pipe Tunnel
2-SE-(-2)-176	Cable Tunnel Access Room

4.2.1.6 Identify Fire Compartments With The Demand For Shutdown

The remaining fire areas and compartments contain systems requiring shutdown either from Technical Specification shutdown initiators or reactor trip initiators. As specified in the FIVE Methodology, these compartments require a Fire Compartment Interaction Analysis to be performed for all barriers.

4.2.1.7 Perform Fire Compartment Interaction Analysis

Fire Compartment Interaction Analyses (FCIAs) were performed to evaluate the likelihood of fire propagation between fire compartment boundaries. These analyses were performed for all significant fire compartments identified in Section 4.2.1.6. The FCIA for each compartment is provided in the Tier 2 documents.

The fire compartment boundaries selected at SONGS 2/3 normally consisted of substantial concrete barriers, many of which are 2 hour rated or equivalent. Each compartment's combustible loading, fire hazard analysis, fire suppression and detection features, and barrier construction was considered in determining the likelihood of a fire propagating beyond the compartment boundaries.

The FCIAs were performed to determine the potential for a fire to spread beyond a single fire compartment. The fire compartments identified in Section 4.2.1.6 resulted in approximately 1000 boundaries. Each boundary was reviewed and, if determined not to represent a potential for fire spread between compartments, was considered an adequate fire compartment boundary. If a boundary was found to be inadequate, the adjacent fire compartments were combined and listed at the end of Section 4.2.1.7. The boundaries which were found to be adequate met one of the following FIVE criteria:

1) Boundaries between two compartments, neither of which contain safe shutdown components nor plant trip initiators, on the basis that fire involving both compartments would have no adverse effect on safe shutdown capability.

2) Boundaries that consist of a 2-hour or 3-hour rated fire barrier. This is based on barrier effectiveness.

3) Boundaries that consist of a 1-hour rated fire barrier with a combustible loading in the exposing compartment <80,000 BTU/ft^2. This is based on barrier effectiveness and combustible loading.

4) Boundaries where the exposing compartment has a very low combustible loading <20,000 BTU/ft^2 and automatic fire detection. Manual suppression will prevent fire spread to the adjacent compartment.

5) Boundaries where both the exposing and exposed compartment have a very low combustible loading <20,000 BTU/ft^2. A significant fire cannot develop in a very low combustible loaded area.

6) Boundaries where automatic fire suppression is installed over combustibles (when combustibles are present) in both the exposing and the exposed compartments. This will prevent fire spread to the adjacent compartment.

Note that FCIA sheets for fire compartments whose boundaries were adequate based on criterion 1 above were not developed. This deviates slightly from the FIVE Methodology which requires an FCIA sheet for all boundaries. A more efficient approach was taken by examining the potential for fire propagation in either direction only across the boundaries of significant fire compartments, whereby significant fire compartments are those that contain equipment which require plant shutdown. This meets the intent of the FIVE Methodology since these are the compartment boundaries of concern. FCIA sheets which verify that the boundaries of significant compartments are adequate to prevent the spread of fire are developed. Also, the analysis performed in response to the FIVE Methodology Revision 1 changes, Section 4.2.1.9 below, validated this screening process.

Information on barriers, combustible loading, suppression, and detection was obtained from UFHA and Appendix R documents.

Plant walkdowns were required for all boundaries which used criteria 3, 4, 5, or 6. This was required by FIVE to ensure there was no concentration of in situ combustibles near the boundary or in situ combustible pathways between compartments. All walkdown documentation is contained in the Tier 2 documents.

Based on these walkdowns and the FCIAs, the following boundaries did not meet one of the criteria above, and therefore may have the potential for fire spread:

- Boundary between 2-AR-50-111A & 2-AR-50-111B
- Boundary between 2-SE-30-142A & 2-CT-(-2)-142B
- Boundary between 2-CT-(-2)-142B & 2-CT-16-142C

- Boundary between 2-TB-9-148A-F & 2-TB-72-154A

- Boundary between 2-YD-30-200A & 2-YD-30-200B

The following fire compartments were therefore combined to form new fire compartments to ensure no fire propagation between compartments. The redefined compartments are as follows:

2-AR-50-111	=	2-AR-50-111A + 2-AR-50-111B
2-SE-CT-142	=	2-SE-30-142A + 2-CT-(-2)-142B + 2-CT-16-142C
2-TB -148	=	2-TB-7-148A + 2-TB-7-148B + 2-TB-9-148C + 2-TB-34-148D + 2-TB-(-9)-148E + 2-TB-9-148F + 2-TB-8-148G + 2-TB-30-148H + 2-TB-72-154A
2-YD-30-200	=	2-YD-30-200A + 2-YD-30-200B

Based on the FCIAs performed above, and the results of Section 4.2.1.6, fire compartments which can be screened from further analysis are listed in Table 4.2-2.

4.2.1.8 Identify Phase II Fire Compartments

All remaining fire compartments which must be analyzed in the Phase II Calculation are those not designated as "screened out" in Table 4.2-2. These are fire compartments which contain systems requiring shutdown, and whose barriers are adequate to prevent the spread of fire.

4.2.1.9 Evaluate FIVE Methodology Revision 1 Changes

On September 29, 1993, NUMARC issued Revision 1 to the FIVE Methodology. Rather than re-performing the completed Phase I analysis for SONGS 2/3, the following evaluation addresses the Revision 1 changes to the FIVE Methodology as they impact the results of this calculation.

Revision 1 of the FIVE Methodology introduced a significant change in the Phase I screening criteria. Revision 0 of FIVE allowed fire areas and fire compartments to be screened from further analysis if they did not contain plant systems requiring shutdown. Revision 1 modified the screening criteria such that a fire area or fire compartment cannot be screened if the area contains plant systems requiring shutdown or if the area contains safe shutdown equipment. Based on this modification to the FIVE Phase I screening criteria, 36 additional SONGS 2/3 fire compartments (as shown in Table 4.2-3) could not be screened.

The approach taken in evaluating these fire compartments involved using the results of the Phase II Calculation A-MN-92-004. By applying conservative values for fire ignition frequency and conditional core damage probability obtained in Phase II for similar fire compartments, bounding core damage frequency (CDF) values were obtained for each of the 36 additional fire compartments. The methodology used to obtain the bounding CDF values, including definitions of the above terms is as follows:

1. The list of safe shutdown components generated in Section 4.2.1.2 above was reviewed in order to identify the safe shutdown equipment losses for each of the 36 fire compartments. For evaluation purposes, fire compartments were categorized based on equipment losses. Fire compartments with similar safe shutdown equipment losses were grouped together as shown in Table 4.2.-3.

2. Fire ignition frequency is the annual probability of a fire occurring in a specific fire compartment and is based on fire compartment location and combustibles. Fire ignition frequencies were assigned to each fire compartment based on values obtained for fire compartments analyzed in Phase II with similar equipment and function. In cases where more than one Phase II fire compartment was applicable, the worst case fire frequency was chosen. The bounding Phase II fire compartments and fire frequencies are provided in Table 4.2-3.

3. Conditional core damage probability (CCDP) is the probability of reactor core damage assuming fire-induced failure of all cables and components within a fire compartment. CCDP values were assigned to each fire compartment based on values obtained for fire compartments analyzed in Phase II with similar safe shutdown equipment losses as defined above. The bounding Phase II fire compartments and CCDP values are provided in Table 4.2-3.

4. Core damage frequency (CDF) is the overall frequency of reactor core damage in events per year obtained by multiplying the fire ignition frequency and the CCDP value. The bounding CDF values for each fire compartment are provided in Table 4.2-3.

5. In accordance with the FIVE Methodology, a fire compartment can be screened from further analysis if the CDF is less than 1E-06/yr. Comparing the CDF bounding values for each of the 36 fire compartments to this screening criterion ensured that the initial Phase I qualitative screening analysis was conservative, and that no further analysis was necessary.

4.2.2 CONTAINMENT AREA FIRE EVALUATION

Fire compartments located in containment (2-CO-15-1A, 2-CO-15-1B, 2-CO-15-1C, and 2-CO-63-1D) were not included in the standard Phase II evaluation process based on guidance provided in the FIVE Methodology. In accordance with the FIVE Methodology, a qualitative assessment was performed in order to determine if the containment needs to be analyzed in the more detailed manner described by FIVE for other plant compartments. For example, FIVE indicates that consideration should be given to conducting an analysis if: (1) the plant experience indicates that fires in containment during power operation are recurrent and (2) redundant trains of critical equipment within containment might be exposed to the same fire plume or be in a confined space and susceptible to damage by a hot gas layer.

SONGS 2/3 fire department records were reviewed to assess fire department responsiveness as discussed in FIVE Section 5.5, Simplified Fire Modeling. Approximately 200 actual fire department incident reports were reviewed covering a five year period. There were no incidents of fires in containment during power operation in this period.

Appendix R documents were reviewed in order to determine if redundant trains of critical equipment might be exposed to the same fire plume or be in a confined space and susceptible to damage by a hot gas layer. In particular, Document 90035AL of UFHA [4-9] discusses the separation of redundant safe shutdown cables and equipment in containment. Relevant information is as follows:

Redundant Systems in Containment	Minimum Separation Distance
Steam Generator Instrumentation (Channel A and Channel B)	30 feet
RCS Temperature Instrumentation (Channel A and Channel B)	130 feet
Containment Emergency Fan Coolers (Train A and Train B)	20 feet
Pressurizer Pressure & Level Instrumentation (Pressure Channels A-D, Level Channels A,B)	17 feet

The separation distances listed above demonstrate that redundant trains of critical equipment within containment cannot be exposed to the same fire plume. Separation between redundant trains is maintained within the confined spaces of containment (i.e., steam generator compartments); therefore, redundant trains of critical equipment are not susceptible to damage by a hot gas layer.

Based on this evaluation, it was determined that the fire compartments in containment are not risk significant. Containment fires during operation were not found to be recurrent, and redundant trains of critical equipment were not found to be susceptible to the same fire plume or hot gas layer. No further analysis was performed for containment compartments.

4.2.3 UNIT 2/UNIT 3 COMPARISON

The Phase I methodology above was performed to screen fire areas and fire compartments with no significant impact on the ability to achieve and maintain shutdown for the Unit 2 reactor. A comparison was performed in order to verify that Unit 3 results would be consistent with Unit 2 results.

The UFHA Technical Specification Barrier Drawings were reviewed to determine the Unit 3 equivalent fire compartment for each of the fire compartments which impact Unit 2. Also, safe shutdown component losses for Unit 3 fire compartments were compared to the safe shutdown component losses for Unit 2 fire compartments.

Based on this review, the following Unit 3 fire compartments required additional consideration based on the unique component failures listed:

Unit 3 Fire Compartment	Unit 2 Fire Compartment	Unique Unit 3 Component Failures
3-PE-45-3A	2-PE-45-3A	Loss of Containment Emergency Cooler Supply and Return valves 3HV-6368, 3HV-6369. Loss of E088 Blowdown Sample Isolation valve 3HV-4057.
3-SE-(-15)-136	2-SE-(-15)-136	Shutdown Cooling Flow to HX E004 Isolation valves 3HV-8152, 3HV-8153. Loss of LPSI Pump 3P-015.

Note that the shutdown procedure for these areas credit manual operation of the valves listed above. These unique Unit 3 areas were considered in the Phase II analysis.

Note that cable routing for Unit 3 non-Appendix R components was assumed to be consistent with the cable routing for Unit 2 non-Appendix R components. This is judged to be acceptable based on the above comparison where almost all of the component and cable losses for Unit 3 were the same as Unit 2.

TABLE 4.2-2
PHASE I SCREENING OF FIRE COMPARTMENTS PER FIVE REV. 0

*(An "x" in the SO column indicates fire compartments which have been screened out. These fire compartments have satisfied the screening criteria and require no further evaluation)

SO*	FIRE COMPARTMENTS	DESCRIPTION
	2-CO-15-1A	Steam Generator Room #2
	2-CO-15-1B	Steam Generator Room #1
	2-CO-15-1C	Containment Quads 1,2,3,4
	2-CO-63-1D	Operator Floor
	2-PE-9-2A	Piping Area
	2-PE-(-18)-2B	Piping Area
	2-PE-30-2C	Piping Area
	2-PE-30-2D	Piping Area
	2-PE-45-3A	Electrical Penetration
	2-PE-63-3B	Electrical Penetration
	2-AC-9-5	Cable Spreading Room
	3-AC-9-6	Cable Spreading Room
	3-AC-9-7	Cable Riser Gallery
X	2-AC-9-8	Lighting Switchgear Room
	2-AC-9-9	Emergency Chiller Room
X	2-AC-9-10	Normal Chiller Room
	2-AC-9-11	Emergency Chiller Room
X	2-AC-9-12	HVAC Room
X	2-AC-9-13	Lighting Switchgear Room
	2-AC-9-14	Cable Riser Gallery
X	2-AC-9-15	Staircase
	2-AC-9-16	Corridor
	2-AC-9-17	Relay Room
X	2-AC-9-18	Elevator
X	2-AC-9-19	Staircase
	2-AC-30-20A	Control Room
X	3-AC-30-20B	Computer Room

SO*	FIRE COMPARTMENTS	DESCRIPTION
X	2-AC-30-20E	Lobby
	3-AC-30-21	Cable Riser Gallery
X	2-AC-30-22	Corridor/Stair
X	2-AC-30-23	Fan Room
X	2-AC-30-24	Staircase
	2-AC-30-26	Fan Room
	2-AC-30-27	Corridor/Stair
	2-AC-30-28	Cable Riser
	2-AC-50-29	Lobby/Motor Control Room
	3-AC-50-30	HVAC Room 3A
X	3-AC-50-31	HVAC Room 3A
	3-AC-50-32	Cable Riser Gallery
	3-AC-50-33	Cable Riser Gallery
	3-AC-50-34	Switchgear Room 3B
	2-AC-50-35	Switchgear Room 2B
	2-AC-50-36	Cable Riser Gallery
	2-AC-50-37	Cable Riser Gallery
	2-AC-50-38	HVAC Room 2A
	2-AC-50-39	HVAC Room 2B
	2-AC-50-40	Switchgear Room 2A
X	2-AC-50-41	Distribution Room
	2-AC-50-42	Battery Room
	2-AC-50-43	Evacuation Room
	2-AC-50-44	Distribution Room 2B
	2-AC-50-45	Distribution Room 2D
	2-AC-50-46	Distribution Room 2C
	2-AC-50-47	Distribution Room 2A

TABLE 4.2-2
PHASE I SCREENING OF FIRE COMPARTMENTS PER FIVE REV. 0

*(An "x" in the SO column indicates fire compartments which have been screened out. These fire compartments have satisfied the screening criteria and require no further evaluation)

SO*	FIRE COMPARTMENTS	DESCRIPTION
	2-AC-30-20C	Computer Room
X	2-AC-39-20D	Technical Support Center
	2-AC-50-51	Battery Room 2B
	3-AC-50-55	Battery Room 3A
	3-AC-50-56	Distribution Room 3A
	3-AC-50-60	Switchgear Room 3A
	2-AC-70-63	Cable Riser Gallery
X	2-AC-70-64	Health Physics
X	2-AC-70-66	HVAC Duct Shaft
X	2-AC-70-67	HVAC Duct Shaft
X	2-AC-70-68	Duct Shaft
X	2-AC-70-69	Duct Shaft
	2-AC-85-70	Switchgear Room
	2-AC-85-71	Switchgear Room
X	2-AC-85-72	Fan Room
X	2-AR-8-73	Primary Plant Makeup Tank
X	2-AR-9-74	Tank Rooms
X	3-AR-9-75	Primary Plant Makeup Tank
	2-AR-9-76	Corridor & Rooms
X	2-AR-9-77	Staircase
X	3-AR-9-78A	Boric Acid Makeup Pump
X	3-AR-9-78B	Boric Acid Makeup Pump
X	2-AR-9-80	Chemical Waste Tank Room
X	2-AR-9-81	Radwaste Primary Tank Room
X	2-AR-9-82	Misc. Waste Evaporator

SO*	FIRE COMPARTMENTS	DESCRIPTION
	2-AC-50-48	Battery Room 2A
	2-AC-50-49	Battery Room 2C
	2-AC-50-50	Battery Room 2D
	2-AR-9-84B	Boric Acid Makeup Pump
X	2-AR-9-86	Staircase
	2-AR-9-87	Charging Pump Room
X	2-AR-9-88	Charging Pump Room
X	2-AR-9-89	Charging Pump Room
X	2-AR-9-90	Elevator
X	3-AR-9-91	Charging Pump Room
X	3-AR-9-92	Charging Pump Room
	2-AR-24-94	Corridor & Rooms
X	3-AR-24-96	Boric Acid Makeup Tank Area
	2-AR-24-98	Boric Acid Makeup Tank Area
X	2-AR-24-99	Duct Shaft Room
X	2-AR-24-100	Letdown Heat Exchanger Room
X	3-AR-24-101	Letdown Heat Exchanger Room
	2-AR-37-102A	Corridor & Rooms
X	2-AR-24-102B	Equipment Room
X	3-AR-37-104	Pipe Room
X	2-AR-37-105	Pipe Room
X	2-AR-37-107	Tank Rooms
X	2-AR-37-108	Rad. Pipe Chase
X	3-AR-37-109	Tank Rooms
X	3-AR-37-110	Rad. Pipe Chase

TABLE 4.2-2
PHASE I SCREENING OF FIRE COMPARTMENTS PER FIVE REV. 0

*(An "x" in the SO column indicates fire compartments which have been screened out. These fire compartments have satisfied the screening criteria and require no further evaluation)

SO*	FIRE COMPARTMENTS	DESCRIPTION
X	2-AR-9-83	Concentrated Boric Acid Tank
X	2-AR-9-84A	Boric Acid Makeup Pump
	2-AR-9-84B	Boric Acid Makeup Pump
X	2-AR-63-121	Tank Room
X	2-FH-17-122	Fuel Tank Room
X	2-FH-17-123	Spent Fuel Pool
X	2-FH-15-124	Staircase
X	2-FH-15-125	Storage Room
X	2-FH-30-126	Heat Exchanger Room
X	2-FH-30-127	Tool Decon. Room
X	2-FH-30-128	Vestibule
X	2-FH-30-129	Dumbwaiter
X	2-FH-45-130	A/C Room #2
X	2-FH-45-131	Vestibule
X	2-FH-45-132	A/C Room #1
X	2-FH-63-134	Vestibule
	2-SE-(-5)-135A	Piping Room/Heat Exchanger
	2-SE-(-5)-135B	Train B CCW Pump Room
	2-SE-(-5)-135C	Spare CCW Pump Room
X	2-SE-(-5)-135D	Train A CCW Pump Room
	2-SE-(-15)-136	Staircase A/C Room
X	2-SE-(-15)-137A	Safety Related Pump Room
X	2-SE-(-15)-137B	Safety Related Pump Room
	2-SE-(-15)-137C	Safety Related Pump Room
X	2-SE-(-15)-138	Heat Exchanger Room
	2-AR-50-111	Electrical Raceway & Rooms
	2-AR-63-116	Corridor & Rooms
	2-AR-63-119	Cable Tray Gallery
X	2-AR-63-120	Duct Shaft Room
	2-SE-CT-142	Electrical Cable Tunnels
X	2-SE-30-143	Elevator
X	2-SE-30-144	Staircase
	2-SE-30-145A	Main Steam Valve Area
X	2-SE-25-145B	AFW Steam Trench
X	2-SE-50-146	Roof-Safety Equipment
	2-TB-148	Turbine Building
X	2-TB-7-149	Main Lube Oil Building
X	2-TB-7-150	Elevator Shaft
	2-TB-30-153	Switchgear Room
	2-TB-72-154B	Accessway
	2-DG-30-155	Diesel Generator Room
X	2-DG-30-156	Staircase
X	2-DG-30-157	Staircase
	2-DG-30-158	Diesel Generator Room
X	2-DG-20-159	Diesel Fuel Transfer Room
X	2-DG-20-160	Diesel Fuel Transfer Room
	2-TK-30-161A	AFW Pump Room
	2-TK-(-2)-161B	AFW Pipe Tunnel
X	2-TK-30-162	Nuclear Service Water Storage
X	2-TK-30-163	Refueling Water Storage Tank

TABLE 4.2-2
PHASE I SCREENING OF FIRE COMPARTMENTS PER FIVE REV. 0

*(An "x" in the SO column indicates fire compartments which have been screened out. These fire compartments have satisfied the screening criteria and require no further evaluation)

SO*	FIRE COMPARTMENTS	DESCRIPTION
X	2-SE-(-15)-139	Heat Exchanger Room
	2-SE-8-140A	Surge Tank Room
	2-SE-8-140B	Chemical Storage Room
X	2-SE-8-141	Surge Tank Room
X	2-AC-(-5)-169	Emergency Chilled Water Pipe
	2-SE-(-12)-170	Emergency Recirculation Tunnel
X	2-SE-30-171	Clean Clothing Room
X	2-FH-30-174A	Railroad Bay
X	2-FH-30-174B	Canopy Area
X	2-AC-70-175	Communicaitons Room
X	2-SE-(-2)-176	Cable Tunnel Access Room
X	2-AR-68-178A	Personnel Facility Area
X	2-AR-68-178B	Hot & Cold Dry Clean Area
X	2-TK-30-165	Condensate Storage Tank

SO*	FIRE COMPARTMENTS	DESCRIPTION
X	2-TK-30-164	Condensate Storage Tank
X	2-TK-30-166	Refueling Water Storage Tank
X	2-CO-15-167	Elevator Shaft
X	2-CO-15-168	Staircase
X	2-AC-85-180	Communications Battery Room
	2-YD-30-200	Unit 2 & Unit 3 Yard Area
	3-SE-CT-142	Electrical Cable Tunnels
	3-TB-148	Turbine Building
	3-TK-30-161A	AFW Pump Room
	3-AR-63-118	Cable Tray Gallery
	3-AC-70-65	Cable Riser Gallery
	3-PE-45-3A	Electrical Penetration Area
	3-PE-63-3B	Electrical Penetration Area/ Personnel Mon.
	3-SE-(-15)-136	Staircase A/C Room

TABLE 4.2-3
FIVE REV. 1 EVALUATION

Fire Compartments Unscreened by FIVE Revion 1		Bounding Fire Ignition Frequency		Bounding Conditional Core Damage Probabiltiy		CDF
Fire Compartment	Description	Fire Compartment	Fire Frequency	Fire Compartment	CCDP	
2-AC-30-24 3-AC-50-31	Staircase HVAC Room 3A	2-AC-30-27 2-AC-50-38	< 1E-3	2-AC-50-39	< 1E-3	< 1E-6
2-AC-9-13 2-AC-(-5)-169 2-AR-9-82 2-TK-30-163 2-TK-30-164	Lighting Switchgear Rm ECW Pipe Tunnel Misc. Waste Evap Cond Monitor Tank Room RWST Room Condensate Storage Tk	2-AC-85-70 2-TK-(-2)-161B 2-SE-8-140A 2-SE-8-140A 2-SE-8-140A	< 1E-2	2-AR-9-84B	< 1E-5	< 1E-7
2-FH-45-130 2-FH-45-132 2-FH-63-134 2-AC-9-12 2-AC-30-22 2-AC-30-23 2-AC-30-20E 2-FH-17-123	AC Room #2 AC Room #1 Vestibule 403 HVAC Room Corridor/Stair Fan Room Lobby SFP/Oper. Floor	2-AC-50-38 2-AC-50-38 2-AC-30-27 2-AC-50-38 2-AC-30-27 2-AC-30-26 2-AC-30-27 2-AC-24-94	< 4E-3	2-AC-9-9	< 2.5E-4	< 1E-6
2-DG-30-157 2-DG-20-159 2-DG-20-160	Staircase D/G Fuel Xfr Pp Rm A D/G Fuel Xfr Pp Rm B	2-AC-30-27 2-SE-(-15)-137C 2-SE-(-15)-137C	< 1.5E-3	2-DG-30-158	< 6.5E-4	< 1E-6

2-AR-24-100 2-AR-63-120 2-AC-50-41 2-AR-9-83 2-AR-9-84A 2-AR-9-88 2-AR-9-89 2-AR-24-99	Letdown HX Room Duct Shaft Room Distribution Room Con Boric Acid Tk Rm BAMU Pump Room Charging Pump Room Charging Pump Room Duct Shaft Room	2-SE-(-5)-135A 2-AC-30-27 2-AC-50-44 2-SE-8-140A 2-SE-(-5)-135C 2-SE-(-5)-135C 2-SE-(-5)-135C 2-AC-30-27	< 1E-2	2-AR-9-87	< 1E-5	< 1E-7
2-AR-24-102B 2-AC-70-64 2-SE-(-15)-138 2-SE-(-15)-139	Equipment Room HP/Access Cntrl Areas Heat Exchanger Room Heat Exchanger Room	2-SE-(-15)-137 2-AR-24-94 2-SE-(-5)-135A 2-SE-(-5)-135A	< 2E-3	2-PE-63-3B	< 5E-4	< 1E-6
2-SE-8-141	Surge Tank Room	2-SE-8-140A	< 1E-3	2-SE-(-5)-135C 2-SE-(-15)-137C	< 1E-4	< 1E-7
2-SE-(-5)-135D 2-AR-9-73	Train A CCW Pump Rm PPMU Tank Rm	2-SE-(-5)-135C 2-SE-(-15)-137C	< 1E-2	2-SE-(-5)-135C	< 1E-4	< 1E-6
2-SE-(-15)-137A 2-SE-(-15)-137B	Sfty Related Pump Rm Sfty Related Pump Rm	2-SE-(-15)-137C 2-SE-(-15)-137C	< 1E-2	2-SE-(-15)-137C	< 1E-4	< 1E-6
2-TK-30-166	RWST	2-SE-8-140A	< 1E-3	2-TK-(-2)-161B	< 1E-3	< 1E-6

4.3 SIMPLIFIED FIRE COMPARTMENT ANALYSIS (FIVE PHASE II)

This section provides a description of the FIVE Phase II simplified quantitative fire compartment analysis for SONGS 2/3. It is organized into the following sections:

4.3.1 FIRE IGNITION FREQUENCIES

The following task was performed to evaluate fire compartment-specific fire ignition frequencies for the Phase II fire compartments shown in Table 4.3-1. Ignition source frequency is the probability of fire occurring in a given fire compartment per year. This value was determined for each fire compartment based on fire compartment location and combustibles.

Fire compartments located in the Containment Building (2-CO-15-1A, 2-CO-15-1B, 2-CO-15-1C, and 2-CO-63-1D) were not included in the standard Phase II evaluation process based on guidance provided in the FIVE Methodology. See Section 4.2.2 for the evaluation of containment fire compartments. Also, per FIVE Section 5.1.7, fire compartments 3-SE-CT-142 and 3-TK-30-161A were able to be screened out based on Phase I criteria (see Reference 4-6, Calculation A-92-MN-003).

The following four-step procedure to determine fire ignition source frequency was provided in the FIVE Methodology and uses FIVE Reference Tables 1.1 and 1.2. For each fire compartment, all data and calculations performed were listed on a Fire Compartment Ignition Source Data Sheet (ISDS). The detailed calculations and ISDS sheets are contained in the Tier 2 documentation.

1. Select a Location - For each fire compartment, the appropriate location (room or building) which best corresponds to the fire compartment characteristics was selected from FIVE Table 1.1, "Weighting Factors For Adjusting Generic Location Fire Frequencies For Application To Plant-Specific Locations."

2. Determine a Weighting Factor for the Location (WF_L) - The weighting factor for each location selected above was then determined from FIVE Table 1.

3. Determine a Weighting Factor for Each Type of Ignition Source (WF_{LS}) - The ignition sources located in each fire compartment were determined using the General Arrangement Drawings. FIVE Table 1.2, "Fire Ignition Sources and Frequencies by Applicable Plant Location," and the Table footnotes were used to determine the weighting factor and fire frequency for each ignition source. FIVE Table 1.2 contains:

   - Plant Location

   - Fire Ignition / Fuel Source

   - Ignition Source Weighting Factor Method

   - Fire Frequency

   Ignition source totals (except pumps) per building required to determine weighting factors were counted for Unit 2 and Unit 3 using SETROUTE. For switchgear and MCCs, individual cubicles, and compartments were considered to be separate electrical panels. When obtaining the total number of electrical panel ignition sources, all electrical panels were assumed to be vented.

   Pump totals were determined using the general arrangement drawings and only included large pumps which could potentially impact safety related equipment or cables. The auxiliary building pump total includes pumps from the (-) 15'-6" and (-) 5'-3" elevations of the safety equipment building (shutdown cooling pumps), the Auxiliary Feedwater pump rooms, and the 9' and 24' elevations of the auxiliary radwaste building (charging pumps and pumps which may impact safety related cables). Pumps in the spent fuel pool cooling area were not included as they would not impact safety related equipment. Pumps on the 63' elevation of the radwaste building were determined via walkdowns not to impact safety related equipment or cables.

   In order to determine the ignition source weighting factor for transient ignition sources, FIVE Footnote D of Table 1.2 was used.

4. Consideration of plant-wide ignition sources - Plant-wide ignition sources were considered as applicable to SONGS as follows:

   - Hydrogen lines were determined to run from the 50' elevation of the Auxiliary Radwaste Building to the Turbine Building via the Penetration Area and Auxiliary Building roofs (elev. 85'). Miscellaneous hydrogen was therefore considered in fire compartments 2-AR-50-111 and 2-TB-148.

- Non-qualified cable was not considered as an ignition source because all safety related cable and most non-safety related cable at SONGS is qualified to IEEE No. 383. Non-qualified cable is only present in the form of vendor-supplied control panel installations and thermocouple cables which are located either in conduit or in non-vented metal control panel enclosures. Non-qualified cable is not routed in cable trays where it would be considered an ignition source.

- Junction boxes were not considered as ignition sources because they are non-vented metal enclosures which contain IEEE qualified cable (similar to non-vented electrical cabinets).

- Hot pipe was not considered as an ignition source because all pipe at SONGS which is more than 140 degrees F is insulated for personnel protection reasons.

Note: The FIVE Methodology states: "In Phase II, values can be estimated using methods other than direct counting, including engineering judgement. Attempt to estimate values within about 25%." Values obtained, although reasonably accurate, were not intended to be exact.

Based on the above information, the fire compartment frequency for each ignition source (F_{if}) was calculated by multiplying:

1. The fire frequency (F_f) for an ignition source present in the fire compartment

2. The weighting factor for the location (WF_L)

3. The weighting factor for that ignition source as calculated (WF_{LS})

The calculation was repeated for each ignition source and the results were summed to obtain the total fire frequency F_1 for that fire compartment. A list of fire frequencies for each fire compartment is contained in Table 4.3-1, Phase II Results. In all cases, the fire frequency value was greater than 1.0E-6/yr; therefore, no fire compartments were screened out based on fire frequency value alone.

4.3.2 PLANT SYSTEMS, HUMAN ERROR, SEQUENCES, AND PLANT RESPONSE

This task was performed to evaluate the conditional core damage probability for the Phase II fire compartments. Conditional core damage probability is the probability of reactor core damage assuming fire-induced failure of all unprotected cables/components within a fire compartment. This task utilized the internal event PRA model (IPE) and considered the manual actions specified in the fire safe shutdown procedures. The detailed methodology, assumptions and results for this task are contained in the Tier 2 documentation. They are summarized below:

- The IPE models were revised to reflect specific fire considerations. For example, if offsite power was lost due to the fire, then it was conservatively assumed to remain unavailable for the 24 hour mission time. Diesel generator mission times were increased from 8 to 24 hours. One air compressor was conservatively assumed unavailable due to lack of cable routing details in the turbine building.

- For each fire compartment or scenario, the equipment which would be impacted in the compartment were reviewed, and appropriate basic events in the IPE model were given a value of "true", which represents guaranteed failure. Note that components that were credited in the Appendix R analysis based on approved exemptions to the Appendix R criteria were not credited for the IPE model.

  That is, unless protected components specifically met the Appendix R criteria, they were assumed to be unavailable. Also note that non-Appendix R components, such as main feedwater, were included in the assessment when available. These modifications are in accordance with the FIVE methodology.

- Depending on the fire scenario and timing, human interactions and operator recovery actions were revised and added in accordance with the plant procedures. These operator actions are summarized in Table 4.3-2. Care was taken to take potential dependencies of multiple operator actions into account.

- The transient (e.g., loss of power conversion system), loss of offsite power, small LOCA, and station blackout event trees were used in the analysis. ATWS, larger LOCAs, secondary side line breaks, and SG tube rupture event trees were not required since the simultaneous occurrence of these events with a fire was determined to be insignificant, as documented in the Tier 2 documents.

- Based on the IPE, there is adequate pressurizer inventory to maintain the RCS at acceptable levels with existing leakage. Therefore, CVCS operation is not critical and is not evaluated in the fire models. Similarly, pressurizer heaters are not required for a safe stable state and were not evaluated. It is recognized that availability of both of these functions would be preferred.

- HVAC requirements and operator actions were re-evaluated during the Phase II screening process. The assumptions made for the CCDP screening calculations are similar to the revised IPE model.

The revised IPE models were used to calculate the CCDP for conservative bounding scenarios for each of the Phase II compartments, with the results shown in Table 4.3-1.

The CCDP values obtained in this section for each of the Phase II fire compartments were applied to the fire frequency values determined above. If the product of these two values was less than 1E-6/yr, then the fire compartment was screened out from further analysis. Note that conservative bounding CCDP calculations were used for many scenarios. That is, if a fire compartment had fire losses that were similar to or enveloped by another fire compartment and the 1E-6/yr screening criterion was met with this bounding calculation, then the compartment was screened out and no further analysis was performed to lower the compartment frequency. Therefore, the CCDPs and screening frequencies in Table 4.3-1 should be viewed as conservative bounding calculations. See Table 4.3-1 for fire compartments which did not screen out. These fire compartments were further analyzed using simplified fire modeling, as discussed in the next section.

CCDP values for the unique Unit 3 areas identified in the Phase I Calculation, 3-PE-45-3A and 3-SE-(-15)-136, are also provided. These fire compartments required additional consideration following Phase I as a result of the Unit 2 versus Unit 3 comparison. These Unit 3 fire compartments contained unique component failures not contained in the comparable Unit 2 fire compartments. Based on the CCDP values obtained in this task, it was determined that the additional component failures did not significantly increase the fire risk beyond the risk associated with the comparable Unit 2 compartments. Therefore, since the Unit 2 compartments were determined to bound these Unit 3 compartments, no further analysis was performed for these compartments.

4.3.3 SIMPLIFIED FIRE MODEL

The overall objective of the simplified fire modeling task was to determine a value for the conditional probability of critical combustible loading damage in a fire compartment

(P_{ccl}). P_{ccl} is the value associated with the conditional probability of system target damage within a fire compartment given a fixed or transient ignition source fire. The P_{ccl} value determined for each fire compartment was based on the following factors:

- Location and amount of fixed ignition sources
- Location of target systems
- Probability of critical exposed transient ignition sources
- Automatic suppression system unavailability
- Manual suppression unavailability

In cases where the information used in the fire scenario worksheets was unable to be verified using plant drawings, plant verification walkdowns were performed. Information typically verified by walkdowns included distances between fixed ignition sources and targets, heights of targets and fixed ignition sources, and electrical panel ventilation. The Phase II walkdown documentation is contained in the Tier 2 documents.

This task was performed for all fire compartments which were not screened out in previous steps. The P_{ccl} value was then multiplied by the fire frequency value and the conditional core damage probability for the fire compartment. If the product of these three values was less than 1.0E-6/yr, then the fire compartment was screened out from further analysis. This was the final criteria used in Phase II in order to determine if a fire compartment could be screened out.

The fire modeling worksheets which document the detailed steps of the process are contained in the Tier 2 documentation and include all equations used in calculating required values. This task corresponds to Sections 6.3.3 through 6.3.7 of the FIVE Methodology. The following is a summary of the steps taken in performing this task.

Step 1. Evaluate the Probability of Fixed Exposure Damage (P_f)

A. Identify Worst Case Target System and Fixed Ignition Source

Selection was based on identifying the system or component loss which was the most significant contributor to conditional core damage probability, as calculated in the previous step. In some cases, the worst case target selection involved the loss of redundant trains of safety systems. The worst case target system was typically found to be offsite power to both trains of essential power. The location of selected target systems within the fire compartment was determined using Appendix R related documentation on cables and raceways associated with safe shutdown components located in the subject fire compartment. These raceways were then located on the tray and conduit plans.

The worst case fixed ignition source was selected based on identifying which fixed ignition source would most likely damage the selected target. Typically, identification considered size and proximity of the ignition source relative to the target. Fixed ignition sources were identified using the equipment layout drawings, the tray and conduit plans, and Appendix R related documents. As necessary, fixed ignition sources were identified and/or verified by performing plant walkdowns. All walkdown documentation is contained in the Tier 2 documents.

In some cases, the subject fire compartment contained no fixed ignition sources. Note that per the FIVE Methodology, IEEE 383 qualified cable was not considered to be a credible fixed ignition source. In addition, unvented electrical panels were not considered as ignition sources.

Only three of the 28 compartments modeled were found by this method not to have damage to the worst case target due to fixed combustibles (2-SE-(-15)-136, 2-AC-50-29, 2-TK-30-161A). The second worst case scenario was evaluated for these 3 compartments, and found to be risk insignificant. An additional nine compartments screened due to the fact that they did not contain fixed ignition sources. Note that 16 of the 28 compartments analyzed in this step did not screen by this method due to damage to the worst-case ignition source.

B. Determine the Applicable Fire Scenario Worksheet

This step reviewed the configuration of the target system and fixed ignition source in order to determine the applicable fire scenario worksheet to be used in performing fire modeling. The fire scenario worksheets were developed using the guidelines presented in the FIVE Methodology. The possible fire scenarios were as follows:

- Target in the Plume - The selected target is located directly above a fixed ignition source (e.g. horizontal cable tray routed above a vented electrical panel).
- Target in the Ceiling Jet - The selected target is located near the ceiling but not directly above a fixed ignition source.
- Target in the Hot Gas Layer - The selected target is not located near the ceiling and is not directly above a fixed ignition source.
- Radiant Exposure to Target - The selected target is located vertically adjacent to a fixed ignition source (e.g., vertical cable tray routed adjacent to a vented electrical panel).

C. *Determine If Fire Causes Target Damage*

Using the steps prescribed in the appropriate fire scenario worksheet, the fire scenario was evaluated to determine if the fire was capable of damaging the target. Determination was first based on the temperature rise and energy release due to the fire. If target damage was possible, the time to target damage was determined based on the radiative and convective heat flux to the target. If the fire compartment contained automatic fire suppression, this value was then compared to the calculated sprinkler actuation time in order to determine if the fire would be suppressed prior to target damage.

D. *Determine Automatic Suppression Unavailability (P_{as})*

If it was found that fire suppression could mitigate fire damage, the unavailability of the fire suppression system was determined. Suppression system unavailability is 2E-2 for wet pipe systems and 5E-2 for pre-action and deluge systems. These values are provided in Reference Table 2 of the FIVE Methodology.

If the fire compartment did not contain automatic suppression, or the target was damaged prior to suppression actuation, then the automatic suppression unavailability was set equal to one.

E. *Determine Manual Suppression Unavailability (P_{ms})*

Fire department records were reviewed in order to determine the manual suppression unavailability. Twenty-nine fire drill records for Procedure SO123-XIII-5 from 10/88 through 4/93 were reviewed as well as 62 actual fire incident reports from 8/88 through 7/93 which were applicable to fire response in the plant protected area. Based on this review, the average fire department response time (fire truck at staging area) was found to be seven minutes. The value obtained for P_{ms} considered detector response time but did not consider fire growth time prior to detection or start time for suppression activities from the staging area.

Target damage was determined to occur in less than seven minutes in all fire scenarios; therefore, credit was not taken for manual suppression. The manual suppression unavailability in all cases was set equal to one.

F. Calculate the probability of fixed exposure damage (P_f)

The product of the automatic and manual suppression unavailabilities was calculated in order to determine the overall probability that target damage would occur due to a fixed ignition source fire.

Step 2. Evaluate the Probability of Transient Combustible Exposure Damage (P_{tc})

The purpose of this step is to determine the risk associated with a fire caused by a transient ignition source. The methodology for this step, as outlined in the FIVE Methodology, was initially used to evaluate the probability of transient combustible exposure damage and is similar to the method used to determine fixed combustible exposure damage. However, based on initial results for the subject fire compartments, the method for this step was simplified as follows:

A. Identify Worst Case Target System

The target system selected in the transient scenario was the system with the highest contributor to core damage. This was typically the same as the worst case target system identified for the fixed ignition source scenario above.

B. Determine If Fire Causes Target Damage

Initial results of simplified fire modeling using one or more 32 gallon trash fire(s) as the transient ignition source found that in most cases, the postulated ignition source was capable of causing target damage. In order to eliminate the arbitrary nature of ignition source selection and associated documentation, worst case target damage due to transient sources was assumed to occur for each fire scenario.

C. Determine Automatic Suppression Unavailability

For each transient fire scenario, automatic suppression unavailability was assumed to be 1.0. This value is based on assuming a large transient ignition source capable of causing target damage prior to suppression system actuation.

D. Determine Manual Suppression Unavailability

For each transient fire scenario, manual suppression unavailability was assumed to be 1.0. This value is based on assuming a large transient ignition source capable of causing target damage in less than seven minutes; therefore, credit was not taken for manual suppression.

E. Calculate the Probability of Fire Damage for Transients (P_{fdt})

The product of the automatic and manual suppression unavailabilities was calculated in order to determine the probability that fire damage would occur for transients. In all cases, this value was set to be 1.0.

Steps F through H consider the likelihood that the transient ignition source was located and exposed in the postulated area.

F. Evaluate Critical Combustible Loading Location (u)

The purpose of this step was to determine a value associated with the postulated location of the transient. Per the FIVE Methodology, the value u is the ratio of the target area (A_s) or radial separation distance (A_{sr}), as applicable, and the net area of the fire compartment. Based on the arbitrary nature of A_s and A_{sr} for a transient fire, the ratio u was conservatively assumed to be equal to 1.0 in all cases except for fire compartments 2-AC-50-29 and 2-TK-30-161A. In these two instances, a conservative value was assigned to A_s.

G. Select Probability of Critical Combustible Loading Being Exposed (p)

This value was selected as 0.1 as defined in the FIVE Methodology. This value can be applied if a transient combustible program exists, as at SONGS.

H. Calculate the Probability That Critical Amounts of Transient Combustibles Are Present Between Inspections (w)

Procedure S0123-XV-4.13 was reviewed to determine the transient inspection frequency for the fire compartment. The frequency was typically found to be 52 inspections per year. Inspection results, located in the SONGS computer records by the procedure number, were also reviewed to determine the number of incidents where inspection found the critical transient limit exceeded. In cases where no incidents occurred, a value of one for the number of violations was assumed in order to prevent an erroneous zero value in the calculation of P_{tc}. These values were used to calculate the probability that critical amounts of transients were present between inspections.

I. Calculate the Probability of Transient Combustible Exposure Damage (P_{tc})

This value was calculated as the product of the probability of fire damage for transients (step e), the ratio associated with the critical combustible loading location (step f), the probability of the combustible being exposed (step g), and the probability that critical amounts of transients are present between inspections (step h).

Step 3. Evaluate the Probability of Critical Combustible Loading Damage (P_{ccl})

This value was calculated as the sum of the probability of fixed exposure damage and the probability of transient combustible exposure damage. This value was the final result of the simplified fire modeling task.

Step 4. Evaluate Revised Compartment Core Damage Frequency Due to Fire

The resulting P_{ccl} and core damage frequency (F3) values from the simplified fire modeling are listed in Table 4.3-1, Phase II Results. The P_{ccl} determined above was then multiplied by the previously calculated core damage frequency value to determine if the subject fire compartment could be screened out. Areas could be screened out if the new CDF value was less than 1.0E-6/yr.

4.3.4 SUMMARY OF RESULTS OF SIMPLIFIED FIRE COMPARTMENT ANALYSIS

Phase II analyzed 85 SONGS 2/3 fire compartments per the FIVE Methodology. Table 4.3-1 provides the results of these calculations, and contains the following information:

- Fire Compartment
- Description
- Fire Frequency (F1): The probability of fire occurring in a given fire compartment per year.
- Conditional Core Damage Probability (CCDP): The probability of core damage given fire-induced component failures for a given fire compartment.
- F2 = F1 . CCDP: This value is calculated to determine if the subject fire compartment can be screened from further analysis.
- Screen [YES/NO]: If F2 is less than 1.0E-6 /yr, the area can be screened per the FIVE Methodology.
- Probability of Critical Combustible Loading Damage (P_{ccl}): The probability of target system damage based on combustibles and fire detection and suppression provisions in a given fire compartment.
- F3 = F2 . Pccl: This value is calculated to determine if the subject fire compartment can be screened from further analysis.
- Screen [YES/NO]: If F3 is less than 1.0E-6/yr, the area can be screened per the FIVE Methodology.

Based on this analysis, 17 fire compartments did not meet the screening criterion for Phase II (core damage frequency less than 1.0E-6 events per reactor year). Fire compartments which did not screen in Phase II are further analyzed in Phase III of this calculation.

TABLE 4.3-1
PHASE II RESULTS

Fire Compartment	Description	F1 = Fire Frequency	CCDP (1E=1.0)	F2 = F1 * CCDP	Screen (F2<1E-6)	P_{CCL}	F3 = F2*P_{CCL}	Screen (F3<1E-6)
2-CO-15-1A	Steam Generator Room #2	Screened Based on Evaluation in Section 4.2.2						
2-CO-15-1B	Steam Generator Room #1	Screened Based on Evaluation in Section 4.2.2						
2-CO-15-1C	Containment Area Quadrants 1-4	Screened Based on Evaluation in Section 4.2.2						
2-CO-15-1D	Operating Floor	Screened Based on Evaluation in Section 4.2.2						
2-PE-9-2A	Piping Area	1.47E-03	1.00E-05	1.47E-08	YES			
2-PE-(-18)-2B	Piping Area	3.34E-04	1.00E-05	3.34E-09	YES			
2-PE-30-2C	Piping Area	6.71E-04	1.30E-05	8.72E-09	YES			
2-PE-30-2D	Piping Area	3.34E-04	1.30E-05	4.34E-09	YES			
2-PE-45-3A	Electrical Penetration	2.54E-03	1.10E-02	2.80E-05	NO	1.00	2.80E-05	NO
2-PE-63-3B	Electrical Penetration	2.54E-03	1.10E-02	2.80E-05	NO	1.00	2.80E-05	NO
2-AC-9-5	Cable Spreading Room	2.69E-05	6.30E-02	1.69E-06	NO	3.80E-03	6.44E-09	YES
2-AC-9-9	Emergency Chiller Room	5.39E-04	1.70E-04	9.16E-08	YES			
2-AC-9-11	Emergency Chiller Room	4.91E-04	1.50E-04	7.37E-08	YES			
2-AC-9-14	Cable Riser Gallerv	2.69E-05	6.30E-02	1.69E-06	NO	3.80E-03	6.44E-09	YES
2-AC-9-16	Corridor	7.45E-03	2.00E-04	1.49E-06	NO	1.00	1.50E-06	NO
2-AC-9-17	Relay Room	7.95E-04	1.20E-02	9.54E-06	NO	1.00	9.54E-06	NO
2-AC-30-20A	Control Room	1.94E-02	6.30E-02	1.22E-03	NO	1.00	1.22E-03	NO
2-AC-30-20C	Computer Room	8.31-E04	6.30E-02	5.24E-05	NO	Evaluated in Phase III		
2-AC-30-26	Fan Room	5.21E-04	2.50E-04	1.30E-07	YES			

TABLE 4.3-1
PHASE II RESULTS

Fire Compartment	Description	F1 = Fire Frequency	CCDP (1E=1.0)	F2 = F1 * CCDP	Screen (F2<1E-6)	P_{CCL}	F3 = F2*P_{CCL}	Screen (F3<1E-6)
2-AC-30-27	Corridor/Stair	3.77E-04	1.30E-04	4.90E-08	YES			
2-AC-30-28	Cable Riser	2.69E-05	6.30E-02	1.69E-06	NO	3.80E-03	6.44E-09	YES
2-AC-50-29	Lobby/Motor Control Room	6.42E-04	5.00E-01	3.21E-04	NO	1.64-03	5.21E-07	YES
2-AC-50-35	Switchgear Room 2B	2.65E-03	2.80E-03	7.11E-06	NO	1.00	7.11E-06	NO
2-AC-50-36	Cable Riser Gallery	2.69E-05	2.00E-03	5.38E-08	YES			
2-AC-50-37	Cable Riser Gallery	2.69E-05	0.12	3.23E-06	NO	3.80E-03	7.87E-06	NO
2-AC-50-38	HVAC Room 2A	4.73E-04	1.00E-04	4.73E-08	YES			
2-AC-50-39	HVAC Room 2B	4.25E-04	8.20E-04	3.49E-07	YES			
2-AC-50-40	Switchgear Room 2A	2.53E-03	3.10E-03	7.87E-06	NO	1.00	7.87E-06	NO
2-AC-50-42	Battery Room	6.76E-04	2.00E-04	1.35E-07	YES			
2-AC-50-43	Evacuation Room	3.71E-04	2.60E-05	9.65E-09	YES			
2-AC-50-44	Distribution Room 2B	7.03E-04	5.00E-04	3.52E-07	YES			
2-AC-50-45	Distribution Room 2D	7.03E-04	5.00E-05	3.52E-08	YES			
2-AC-50-46	Distribution Room 2C	7.03E-04	1.001E-05	7.03E-09	YES			
2-AC-50-47	Distribution Room 2A	7.03E-04	6.00E-04	4.22E-07	YES			
2-AC-50-48	Battery Room 2A	6.76E-04	1.00E-05	6.76E-09	YES			
2-AC-50-49	Battery Room 2C	6.76E-04	1.00E-05	6.76E-09	YES			
2-AC-50-50	Battery Room 2D	6.76E-04	4.00E-05	2.70E-08	YES			
2-AC-50-51	Battery Room 2B	6.76E-04	1.00E-05	6.76E-09	YES			

TABLE 4.3-1
PHASE II RESULTS

Fire Compartment	Description	F1 = Fire Frequency	CCDP (1E=1.0)	F2 = F1 * CCDP	Screen (F2<1E-6)	P_{CCL}	F3 = F2*P_{CCL}	Screen (F3<1E-6)
2-AC-70-63	Cable Riser Gallery	3.04E-05	3.70E-02	1.12E-06	NO	3.8E-03	4.27E-09	YES
2-AC-85-70	Switchgear Room	2.53E-03	1.10E-02	2.78E-05	NO	1.00	2.78E-05	NO
2-AC-85-71	Switchgear Room	2.53E-03	1.10E-02	2.78E-05	NO	1.00	2.78E-05	NO
2-AR-9-76	Corridor & Rooms	1.18E-02	7.00E-05	8.26E-07	YES			
2-AR-9-84B	Boric Acid Makeup Pump	9.51E-04	5.50E-06	5.23E-09	YES			
2-AR-9-87	Charging Pump Room	9.51E-04	5.50E-06	5.23E-09	YES			
2-AR-24-94	Corridor & Rooms	1.56E-03	7.00E-05	1.09E-07	YES			
2-AR-24-98	Boric Acid Makeup Tank Area	3.34E-04	5.50E-06	1.84E-09	YES			
2-AR 37-102A	Corridor & Rooms	1.19E-02	7.00E-05	8.33E-07	YES			
2-AR-50-111	Electric Equipment/Raceway	2.42E-03	7.00E-05	1.69E-07	YES			
2-AR-63-116	Corridor & Rooms	1.53E-03	1.10E-02	1.68E-05	NO	1.00	1.68E-05	NO
2-AR-63-119	Cable Tray Gallery	2.69E-05	1.10E-02	2.96E-07	YES			
2-SE-(-5)-135A	Piping/Heat Exchanger Room	1.52E-03	3.00E-03	4.56E-06	NO	3.80E-03	1.73E-08	YES
2-SE-(-5)-135B	Train B CCW Pump Room	1.00E-03	4.00E-04	4.00E-07	YES			
2-SE-(-5)-135C	Spare CCW Pump Room	1.05E-03	3.00E-05	3.15E-08	YES			
2-SE-(-15)-136	Staircase A/C Room	1.56E-03	3.00E-03	4.68E-06	NO	3.80E-03	1.78E-08	YES
2-SE-(-15)-137C	Safety Related Pump Room	2.14E-03	1.10E-05	2.35E-08	YES			
2-SE-8-140A	Surge Tank Room	3.34E-04	5.00E-05	1.67E-08	YES			
2-SE-8-140B	Chemical Storage Room	1.47E-03	5.00E-05	7.35E-08	YES			

TABLE 4.3-1
PHASE II RESULTS

Fire Compartment	Description	F1 = Fire Frequency	CCDP (1E=1.0)	F2 = F1 * CCDP	Screen (F2<1E-6)	P_{CCL}	F3 = F2*P_{CCL}	Screen (F3<1E-6)
2-SE-CT-142	Electrical Cable Tunnels	3.77E-04	5.00E-01	1.89E-04	NO	3.80E-03	7.16E-07	YES
2-SE-30-145A	Main Steam Valve Area	1.02E-03	1.00E-04	1.02E-07	YES			
2-TB-148	Turbine Building	4.51E-02	2.25E-02	1.01E-03	NO	1.04	1.10E-03	NO
2-TB-30-153	Switchgear Room	2.53E-03	1.00E-04	2.53E-07	YES			
2-TB-72-154B	Access way	1.78E-04	1.00E-04	3.56E-08	YES			
2-DG-30-155	Diesel Generator Room	2.87E-02	5.00E-04	1.44E-05	NO	1.0	1.44E-05	NO
2-DG-30-158	Diesel Generator Room	2.87E-02	6.00E-04	1.72E-05	NO	1.0	1.72E-05	NO
2-TK-30-161A	AFW Pump Room	4.62E-03	9.50E-02	4.39E-04	NO	1.47E-03	6.47E-07	YES
2-TK-(-2)-161B	AFW Pipe Tunnel	3.34E-04	1.30E-04	4.34E-08	YES			
2-SE-(-12)-170	Emergency Recirc. Tunnel	3.34E-04	1.00E-04	3.34E-08	YES			
2-YD-30-200	Unit 2 and 3 Yard Area	1.65E-02	1.00	1.65E-02	NO	1.05	1.73E-02	NO
3-PE-45-3A	Electrical Penetration	2.54E-03	1.10E-02	2.79E-05	NO	1.00	2.80E-05	NO
3-PE-63-3B	Electrical Penetration	2.54E-03	1.10E-02	2.79E-05	NO	1.00	2.80E-05	NO
3-AC-9-6	Cable Spreading Room	2.69E-05	6.30E-02	1.69E-06	NO	3.80E-03	6.44E-09	YES
3-AC-9-7	Cable Riser Gallery	2.69E-05	4.00E-02	1.08E-06	NO	3.80E-03	4.09E-09	YES
3-AC-30-21	Cable Riser Gallery	2.69E-05	3.00E-02	8.07E-07	YES			
3-AC-50-30	HVAC Room 3B	4.25E-04	5.00E-05	2.13E-08	YES			
3-AC-50-32	Cable Riser Gallery	2.69E-05	1.20E-02	3.23E-07	YES			
3-AC-50-33	Cable Riser Gallery	2.69E-05	3.40E-05	9.15E-10	YES			

TABLE 4.3-1
PHASE II RESULTS

Fire Compartment	Description	F1 = Fire Frequency	CCDP (1E=1.0)	F2 = F1 * CCDP	Screen (F2<1E-6)	P_{CCI}	F3 = F2*P_{CCI}	Screen (F3<1E-6)
3-AC-50-34	Switchgear Room 3B	2.54E-03	1.50E-04	3.81E-07	YES			
3-AC-50-55	Battery Room 3A	6.76E-04	1.70E-04	1.15E-07	YES			
3-AC-50-56	Distribution Room 3A	7.03E-04	1.70E-04	1.20E-07	YES			
3-AC-50-60	Switchgear Room 3A	2.54E-03	1.70E-04	4.32E-07	YES			
3-AC-70-65	Cable Riser Gallery	2.69E-05	1.10E-02	2.96E-07	YES			
3-AR-63-118	Cable Tray Gallery	2.69E-05	1.10E-02	2.96E-07	YES			
3-SE-CT-142	Electrical Cable Tunnels	Screened Based on Circuit Analysis						
3-TB-148	Turbine Building	4.51E-02	1.00E-05	4.51E-07	YES			
3-TK-30-161A	AFW Pump Room	Screened Based on Circuit Analysis						
3-SE-(-15)-36	Staircase A/C Room	1.56E-03	3.00E-03	4.68E-06	NO	3.80E-03	1.78E-08	YES

TABLE 4.3-2
SUMMARY OF OPERATOR ACTIONS USED IN FIRE IPEEE SCENARIOS

OPERATOR ACTION	FIRE AREA	HUMAN ERROR PROBABILITY IPE	SCREENING	DETAILED
1. Operator fails to manually operate T/D AFWP given loss of DC power	several	6E-3	-	-
2. Operator fails to manually control Train A AFW /HPSI /containment spray from SWGR following a Control Room fire	2-AC-30-20A	-	-	4E-2
3. Operator fails to control Train B AFW system from SWGR following failure of Train A in a Control Room fire	2-AC-30-20A	-	0.1	-
4. Operator fails to manually control T/D AFW valves following loss of trains A & B in a Control Room fire	2-AC-30-20A	-	0.2	-
5. Operator fails to manually control Train B HPSI/ containment spray from SWGR following failure of train A in a Control Room fire	2-AC-30-20A	-	0.2	-
6. Operator fails to manually start EDG in DG room	several	-	-	8E-3
7. Operator fails to activate fire procedures (SO23-13-2)	several	-	-	3E-5
8. Operator fails to crosstie 480 vac buses between units 2/3 within a. 2.5 hrs b. 5 hrs c. 8 hrs	 several (SBO) several (SBO) several (SBO)	 - - -	 0.03 (Note 1) 0.01 (Note 1) 0.01 (Note 1)	 - - -
9. Operator fails to remotely start SWC pumps in Control Room given failure of SWC auto control	2-SE-(-5)-135C	-	1E-3 (Note 2)	-
10. Operator fails to control CCW pumps and non-critical loop after seal LOCA	2-SE-8-140A/B	-	1E-3 (Note 2)	-
11. Operator fails to reopen MSIV and regain MFW from Control Room after spurious closure of MSIV/MFIV (60 min to steam generator dryout)	2-TK-30-161A	0.43	0.8	-
12. Operator fails to isolate faulted line from Reserve Auxiliary Transformer and reconnect offsite power (POS17) to both units in 8 hrs	Containment Penetration Areas	-	0.1	-
13. Operator mistakenly disconnects offsite power from 4kV bus (assumes change in procedure SO23-13-21)	several (e.g., 2-DG-30-158)	-	0.1	-
14. Operator fails to prevent CCW pump runout	several	-	0.1	-

TABLE 4.3-2
SUMMARY OF OPERATOR ACTIONS USED IN FIRE IPEEE SCENARIOS

OPERATOR ACTION	FIRE AREA	HUMAN ERROR PROBABILITY IPE	SCREENING	DETAILED
15. Operator fails to manually trip RCPs if CCW lost to RCP seals or coolers (e.g., spurious closure of non-critical CCW loop valves)	several	5.2E-4	-	-
16. Operator fails to start diesel generator from control room	2-AC-9-17	-	-	3E-3
17. Operator fails to respond to battery charger B2 failure given previous response failed	several	-	0.2 (Note 3)	-
18. Operator fails to align backup battery charger	several	0.1	-	-
19. Operator fails to recover power after loss of offsite power to both units	several	-	0.1 (Note 4)	-

Note 1: The screening values for operator cross-tie 480V buses between opposite units was conservatively derived from the Human Cognitive Reliability (HCR) model for time available at 2.5 hours and 5 hours, respectively. Note that Human Error Probabilities (HEPs) in the fire-induced station blackout event tree bound probability values of the knowledge based HCR model (the most conservative action category). It should be pointed out that these values are lower than that of the IPE station blackout event tree; this is due to the fire event tree is based on 480V buses cross-tie while the IPE model is based on 4KV buses cross-tie. This difference in probability values is directly related to the availability of plant specific procedure for the associated operator actions; the 480V cross-tie procedures has been developed and is readily available (EOI procedures SO23-12-8 "Station Blackout" Attachment 17 and SO23-12-9 "Functional Recovery" Attachment 32, and maintenance procedure SO23-I-4.78), while there is no formal 4KV cross-tie procedure during preparation of the SONGS IPE submittal.

Note 2: Proceduralized operator actions inside control room are provided in AOI SO23-13-21. Sufficient response time for manual control of SWC or CCW pumps to maintain heat removal of ESF components.

Note 3: Charger failure alarm will be actuated after ESF high temperature alarm. Based on the alarm set points, there should be sufficient delay time (about an hour) between high temperature alarm and the charger alarm. Operator is not likely to be confused by these alarms.

Note 4: This action is to restore offsite power after loss of offsite power to both units 2 and 3. After the power loss, it is assumed that in the first eight hours power will be provided by emergency diesels; from the ninth hour the probability of operator fails to recover offsite power is conservatively set to 0.1. This operator action is applicable to fire scenarios: 2-PE-45-3A, 2-PE-63-3B, 3-PE-45-3A, loss of electrical panel CR-63 in a MCR fire, etc.

Using the recovery of Unit 3 offsite power as an example, the required operator actions include isolating 3XR3 by opening its disconnect switch at the transformer with a fiber glass pole. Then re-energize transformers 3XR1 and 3XR2 by reclosing position 17 breakers from the control room. This restores power to class 1E 4KV switchgear. The required action is part of the standard steps for recovery of offsite power; generic procedures are available for these steps and operators are trained on these actions.

4.4 DETAILED FIRE COMPARTMENT ANALYSIS (FIVE PHASE III)

This section contains the Phase III evaluation of the fire compartments which remained unscreened after the simplified fire modeling performed in the Phase II tasks. It is organized into the following sections:

4.4.1 Methods for Detailed Analysis
4.4.2 Compartment Evaluations
4.4.3 Results of Phase III Evaluations

4.4.1 METHODS FOR DETAILED ANALYSIS

Phase III was performed for each of the fire compartments which did not screen out in Phase II. Each fire compartment was assessed individually, in order to evaluate significant contributors to fire risk, and to use the most appropriate means available to determine a realistic estimate of core damage frequency. Phase III fire compartments were evaluated using one or more of the following means:

- COMPBRN Fire Modeling: COMPBRN IIIE [4-2, 4-10], an interactive computer code, has been used extensively in probabilistic risk assessments to evaluate fire growth and propagation in nuclear power plants, and has been approved for use in this application by the NRC. The COMPBRN computer code uses thermal property values and geometric configuration to model the physics of fire growth within a fire compartment and determine the probability of target damage. The COMPBRN IIIE approach to determining the probability of target damage is more detailed than the approach used in simplified fire modeling discussed in Phase II in that it involves the actual application of fire growth theory. The value obtained for target damage probability can be used to reduce the overall core damage frequency for a fire compartment.

- Consideration of administrative changes: In some cases, changes to plant procedures can be effective in reducing the overall core damage probability. Typically, this involves enhancing procedures to include instruction on system recovery, or clarifying procedures to allow for latitude in the performance of certain steps which may reduce risk.

- Fault tree modeling: This process involves dividing a fire compartment into several separate fire scenarios, evaluating each scenario, then combining the results to obtain an overall core damage frequency. Adequate physical separation between scenarios allows the fire frequency and CCDP values to be revised to better represent credible ignition sources for a given target and more accurate CCDP based on specific target loss.

- Evaluation and justification: In some cases, evaluation of specific fire-induced component losses and the shutdown methodology for a fire compartment may be successful in reducing the core damage frequency. CDF reduction can be attributed primarily to reducing conservatism in the Appendix R safe shutdown analysis results for the fire compartment.

The use of COMPBRN IIIE for applicable Phase III fire compartments was standard with the following exceptions:

1. All thermal property values used in the models (e.g., cable) were obtained from the COMPBRN internal database (with the exception of electrical cabinet ignition sources) and are indicative of the types of materials found at SONGS. Values for cable tray mass varied with values up to 8 kg per element. However, no cable trays ignited during the COMPBRN modeling, so this input value did not impact the results.

   For electrical cabinets, a new fuel type (CABINET) was defined. This fuel type used the thermal properties of cable with a decreased damage temperature (400°F)and an increased combustion efficiency. In some models, the variability factor for fuel-surface controlled burning rate was adjusted in order to simulate the heat release rate curve for electrical cabinets (Ref. FIVE Methodology, Figure 5a, page 10.4-44). This was done in cases where room geometry resulted in a heat release rate which was significantly lower than the median heat release rate (approximately 1000 KW). Individual cabinets were generally assigned a mass of 15 kg, however, the mass of the fire source was increased to 125 kg to aid in achieving the desired fire duration.

2. Room geometry was modeled without considering room boundaries. The lack of a hot gas layer does not affect the results as most rooms modeled were very large (penetration areas and control room) or have openings in the ceiling to the atmosphere (2-AC-85-70 and 2-AC-85-71) which would inhibit the formation of a hot gas layer.

   A sensitivity analysis was performed for one remaining area, 2-AC-9-17, in which room boundaries were modeled, and a heat release rate of 100 KW for the cabinet fire was used. This value is given as an upper bound for electrical cabinet heat release rates in the EPRI Fire PRA Implementation Guide. The sensitivity analysis produced less conservative damage probabilities than those calculated for the COMPBRN analyses for SONGS. It is therefore determined that the method used for SONGS is conservative, due to the fact that the heat release rate used in the COMPBRN models (item 1) is ten times the value recommended for electrical cabinets.

3. A probability of non-suppression factor is used in fire PRA to reduce the probability of core damage. COMPBRN was designed to aid in the calculation of the probability of non-suppression (P_{NS}) by providing a value for the mean time to damage. This calculation is usually performed for scenarios in which the probability of damage (P_{dam}) is approximately equal to 1.0.

   In this analysis, P_{dam} was typically found to be much less than 1.0, based on the type of electrical cabinet fires modeled. Therefore, P_{NS} was calculated by multiplication of P_{dam} and the frequency of non-suppression (F_{NS}) which is derived from the mean time to damage. This was necessary to account for the fact that the calculated mean time to damage only applied to the fraction of cases in which damage occurred. For the remainder of cases in which damage does not occur, P_{NS} is zero, because F_{NS} approaches zero when the damage time becomes large. When suppression was not credited, F_{NS} was set equal to 1.0; therefore, P_{NS} became equal to P_{dam}.

   All geometric data was obtained by plant walkdowns. Walkdown documentation and documentation of COMPBRN results are contained in the Tier 2 documents. Room boundary dimensions are within 6% of actual values, which does not impact the results.

4.4.2 COMPARTMENT EVALUATIONS

This section contains the Phase III fire compartment evaluations which were performed for each of the fire compartments that did not screen out in Phase II. Each fire compartment evaluation discusses the following:

- Fire compartment description: This includes, as necessary, fire compartment size and geometry, location and concentration of fixed combustibles, and location of primary contributors to core damage frequency. Note that fires from transient combustibles were considered for all of these compartments, but were shown to be negligible contributors to core damage frequency.

- Scenario development: This section discusses the development of scenarios for the subject fire compartment. These scenarios could then include COMPBRN fire modeling, consideration of administrative changes, fault tree modeling, or evaluation and justification for no further analysis. Note that the conditional core damage probabilities (CCDPs) were developed using the modified IPE models, and vary depending both on the specific scenario losses and the compartment losses. For example, a loss of one emergency bus in one compartment may have a different CCDP than in another compartment due to other losses in the specific compartments.

- Overall Compartment Results: This includes the overall core damage frequency for the subject fire compartment after completion of the Phase III effort. Recommendations for any plant administrative changes applicable to improved CDF values are discussed. Fire compartments which still do not meet the screening criteria will then undergo further analysis of containment and heat removal functions using the IPE Level 2 results.

The combined results for all Phase III evaluations are presented in Section 4.4.3.

4.4.2.1 Fire Compartment 2-PE-45-3A, Electrical Penetration Room (Penetration Building 45')

In order to better estimate the core damage frequency for this fire compartment, five separate fire scenarios were evaluated: a small fixed ignition source fire, three large fixed ignition source fires, and a transient ignition source fire. The results for each of these scenarios are shown in Table 4.4-1. The initiating fire frequencies of the small fire scenario and large fire scenarios were based on a categorization of fires contained in the fire frequency database for switchgear fires, combined with the appropriate weighting factors as previously described. This fire compartment was evaluated as a switchgear room because it contains 6.9 kV switchgear and associated cables, 480 V transformers and cabinets. The switchgear room fires in the Fire Events Database [4-11] were divided into eight small fires (incident numbers 214, 310, 324, 475, 494, 534, 663, and 671) and eight large fires (incident numbers 65, 127, 175, 195, 349, 498, 516, and 642). Three events were determined to be not applicable (incident numbers 173, 221, 634) because two were due to personnel errors during outage activities and the other cannot occur at SONGS Units 2 and 3 because of the switchgear room construction. Figure 4.4-1 is a layout drawing depicting this compartment.

The small fire scenario consists of fire events which are 5 minutes [illegible] in duration and are confined to the originating cabinet with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, because loss of a single breaker cabinet could not cause loss of both POS4 and POS17 (offsite power to Unit 3 essential power). Therefore, the fire frequency for small fires in this compartment was conservatively multiplied with the conditional CDP for loss of POS4, to obtain the small fire core damage frequency (CDF) of 1.06E-7/yr.

TABLE 4.4-1
FIRE COMPARTMENT 2-PE-45-3A, ELECTRICAL PENETRATION ROOM

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.00E-4	1.06E-7
Large fire with loss of POS4 and POS17	3.53E-4	1.0	2.9E-3	1.02E-6
Large fire with loss of 2A04 and POS4	2.94E-4	0.31	2.8E-3	2.55E-7
Large fire with loss of POS4	6.23E-4	1.0	2.0E-4	1.25E-7
Transient fire with loss of POS4, POS17	3.14E-5	3.8E-3	1.1E-2	1.31E-9
COMPARTMENT TOTAL				1.5 E-6

The large fire scenarios consist of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. Three separate large fire scenarios are evaluated: (1) loss of POS4 and POS17 due to a fire in switchgear 2A01, (2) loss of emergency switchgear 2A04 due to a fire in 2B02, and (3) procedural loss of POS4 due to a fire in the remaining fixed ignition sources. All of these scenarios assume a procedural (SO23-13-21, "Fire") trip of POS4 to mitigate spurious actuations, due to the inaccessibility of 2A01 in the presence of a large fire in this compartment. The results of these scenarios are added to obtain the total CDF for large fires.

The primary contributor to core damage frequency in this fire compartment is the loss of offsite power (POS4) to Unit 2 essential power in conjunction with the loss of offsite power (POS17) to Unit 3 essential power. COMPBRN modeling determined that damage to both POS4 and POS17 could occur only from switchgear 2A01. Based on the fire frequency of 2A01, and the CCDP associated with losses of POS4 and POS17, the scenario CDF was 1.02E-6/yr.

The secondary contributor to core damage frequency in this compartment is the loss of train A essential switchgear 2A04. Fire-induced failure of the cables associated with 2A04 results in the inability to power 2A04 from the diesel, and inadvertently trips 2A04 from the offsite power source, (POS4). Using COMPBRN modeling, loss of 2A04 was determined to occur only from bus 2B02. Loss of 2A04, and procedural loss of POS4 resulted in a CDF of 2.55E-7/yr.

The remaining fixed ignition sources in this compartment are conservatively assumed to cause damage to all safe shutdown cables in the compartment, except those evaluated in the previous scenarios. Assuming the fire is large enough to prevent access to switchgear 2A01, the component losses would consist of various CCW, CVCS and RCS components, as well as a procedural trip of POS4. The resulting CCDP was multiplied by the fire frequency of the remaining fixed ignition sources, including electrical panels and HVAC units to obtain a CDF of 1.25E-7/yr for this scenario. The fire frequency also includes the fraction of fire frequency for bus 2B02 that was not used in the second large fire scenario.

A worst case scenario for this fire compartment would be loss of both POS4 and POS17 in conjunction with loss of 2A04, leaving only diesel-powered train B components for shutdown. However, this was determined not to be a credible scenario using COMPBRN modeling, which determined that the 2A04 raceway CYANA7 could not be damaged by a fire in switchgear 2A01, from which it is separated by about 11 feet.

The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was multiplied by the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 1.5E-6/yr. The fire compartment will be evaluated for containment heat removal and isolation capability (as discussed in Section 4.5) using the results of the IPE Level 2 Analysis.

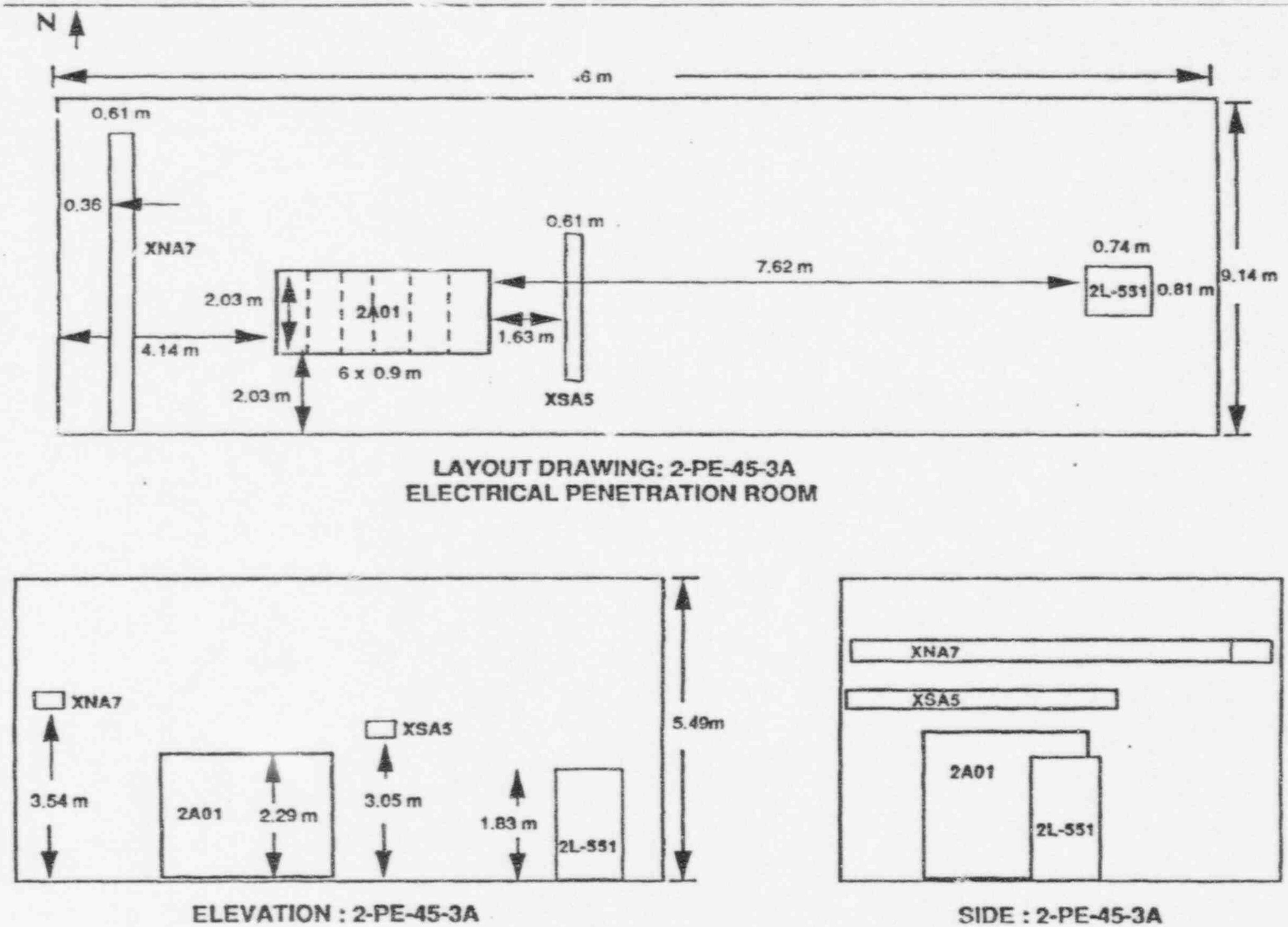


FIGURE 4.4-1

COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-PE-45-3A

4.4.2.2 Fire Compartment 2-PE-63-3B, Electrical Penetration Room (Penetration Building 63')

This compartment evaluation is similar to the previous compartment, 2-PE-45-3A. Five separate fire scenarios were evaluated: a small fixed ignition source fire, three large fixed ignition source fires, and a transient ignition source fire. The results for each of these scenarios are shown in Table 4.4-2. The initiating fire frequencies of the small fire scenario and large fire scenarios were based on a categorization of fires contained in the fire frequency database for switchgear fires, and use of the appropriate weighting factors as previously described. This fire compartment was evaluated as a switchgear room because it contains 6.9 kV switchgear and associated cables, 480 V transformers and cabinets. Figure 4.4-2 is a layout drawing depicting this compartment.

TABLE 4.4-2
FIRE COMPARTMENT 2-PE-63-3B, ELECTRICAL PENETRATION ROOM

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.00E-4	1.06E-7
Large fire with loss of POS4 and POS17	6.48E-4	1.0	2.9E-3	1.88E-6
Large fire with loss of POS17, and POS4 trip	4.12E-4	0.31	2.9E-3	3.70E-7
Large fire with loss of POS4	2.88E-4	1.0	2.0E-4	5.77E-8
Transient fire with loss of POS4, POS17	3.14E-5	3.8E-3	1.1E-2	1.31E-9
COMPARTMENT TOTAL				2.4 E-6

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, because loss of a single breaker cabinet could not cause loss of both POS4 and POS17 (offsite power to Unit 3 essential power). Therefore, the fire frequency for small fires in this compartment was conservatively multiplied with the conditional CDP for loss of POS4, to obtain the small fire core damage frequency (CDF) of 1.06E-7/yr.

The large fire scenarios consist of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. Three separate large fire scenarios are evaluated, (1) loss of POS4 and POS17 due to a fire in switchgear 2A02 or bus 2B08, (2) loss of POS17 due to a fire in the remaining fixed ignition sources, with procedural trip or loss of POS4, and (3) procedural loss of POS4 due to a fire in the remaining fixed ignition sources. All of these scenarios assume a procedural

(SO23-13-21, "Fire") trip of POS4 to mitigate spurious actuations, due to the inaccessibility of 2A02 in the presence of a large fire in this compartment. The results of these scenarios are added to obtain the total CDF for large fires.

The primary contributor to core damage frequency in this fire compartment is the loss of offsite power (POS4) to Unit 2 essential power in conjunction with the loss of offsite power (POS17) to Unit 3 essential power. COMPBRN modeling determined that damage to both POS4 and POS17 could occur from a large fire in switchgear 2A02 or from a fire damaging cable raceways above bus 2B08. Based on the fire frequency of 2A02 and 2B08, and the CCDP associated with losses of POS4 and POS17, the scenario CDF was 1.88E-6/yr.

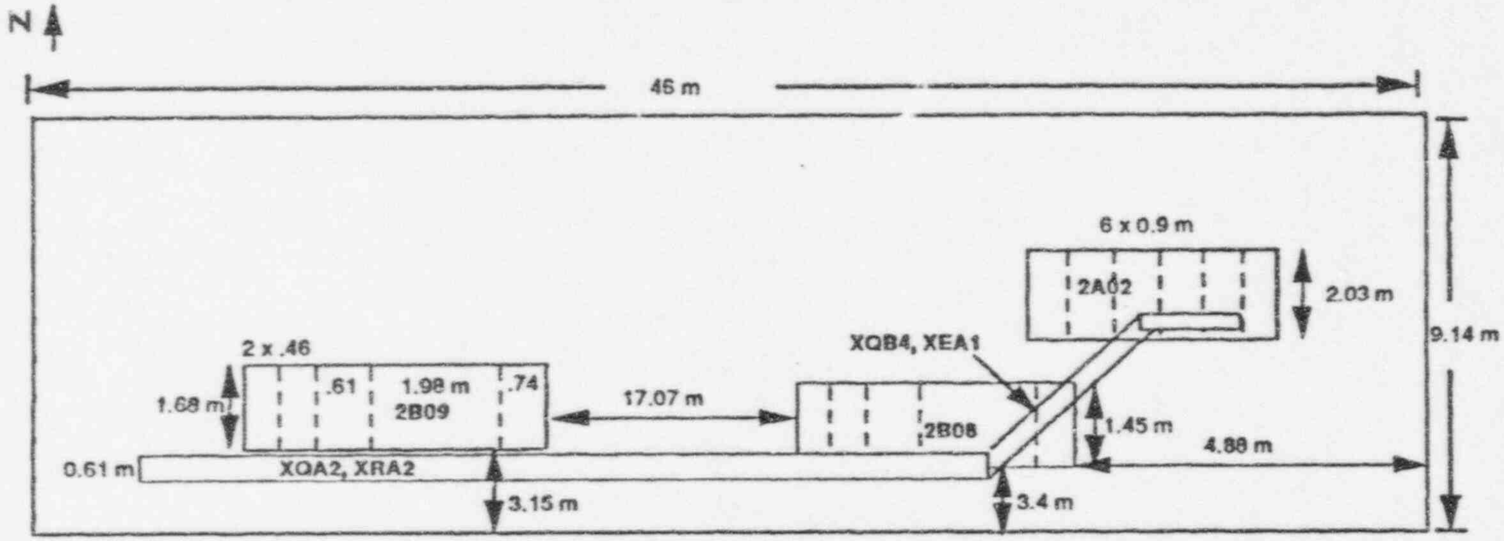
The secondary contributor to core damage frequency in this compartment is the loss of POS17 due to a fire in the remaining fixed sources, and the procedural trip or fire loss of POS4. Using COMPBRN modeling for the worst case fire source and raceway, fire in 2B09 was determined to spread to POS17 cables in raceway XQA2 with a probability of 0.31. Loss of POS17 and procedural trip or loss of POS4 resulted in a CDF of 3.7E-7/yr.

The remaining fixed ignition sources in this compartment are conservatively assumed to cause damage to all safe shutdown cables in the compartment, except those evaluated in the previous scenarios. These component losses consist of various CCW, CVCS and RCS components, as well as a procedural trip of POS4, assuming the fire is large enough to prevent access to switchgear 2A02,. The resulting CCDP was multiplied by the fraction of the fire frequency of the remaining fixed ignition sources not used in the previous scenario, (including electrical panels and an HVAC unit), to obtain a CDF of 5.7E-8/yr for this scenario.

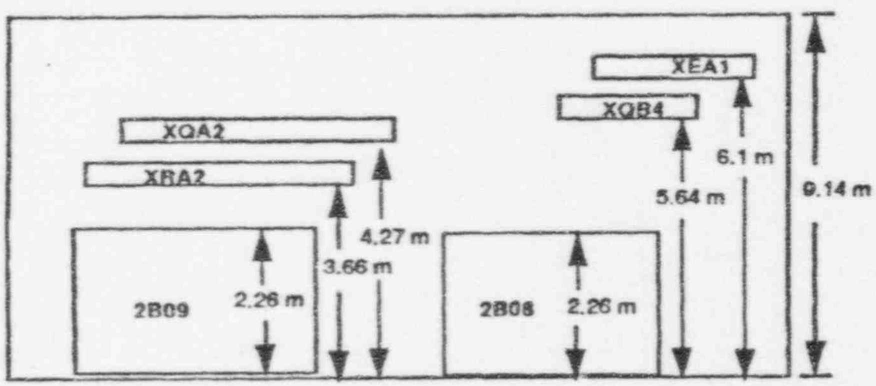
The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was multiplied by applied to the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

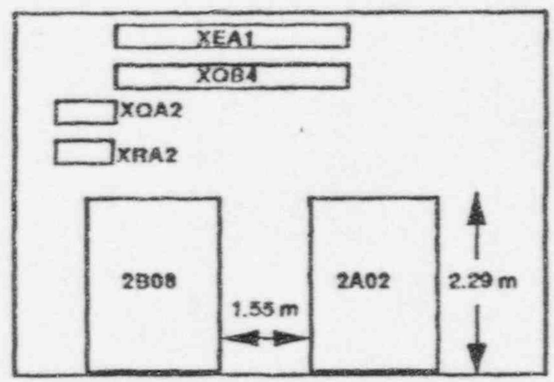
The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 2.4E-6/yr. The fire compartment will be evaluated for containment



LAYOUT DRAWING: 2-PE-63-3B
ELECTRICAL PENETRATION ROOM



ELEVATION : 2-PE-63-3B



SIDE : 2-PE-63-3B

FIGURE 4.4-2
COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-PE-63-3B

heat removal and isolation capability (as discussed in Section 4.5) using the results of the IPE Level 2 Analysis.

4.4.2.3 Fire Compartment 2-AC-9-16, Corridor (Auxiliary Control Building, 9')

Fire compartment 2-AC-9-16 is the corridor located on the 9' elevation of the Auxiliary Control Building. The primary contributor to core damage frequency in this fire compartment is the loss of Train B Emergency Chilled Water (ECW). Circuitry for this system is routed in the corridor portions of this fire compartment with some circuitry terminating in panel 2/3L-378. Based on the location and configuration of Train B ECW circuitry relative to fixed ignition sources within the fire compartment, the 4 fire scenarios below were considered. In each case, revised fire frequency values and CCDP values were applied to the scenarios to better represent credible ignition sources for a given target and more accurate CCDP values based on specific target loss. The overall CDF values for the following four fire scenarios were then added to obtain a revised overall CDF value, as shown in Table 4.4-3:

Scenario 1: Loss of Train B ECW due to 2/3L-378 fire
Scenario 2: Loss of Train B ECW due to remaining fixed ignition sources (automatic suppression unavailable)
Scenario 3: No significant component losses due to remaining fixed ignition sources (automatic suppression available)
Scenario 4: Loss of Train B ECW due to transient combustible fires

TABLE 4.4-3
FIRE COMPARTMENT 2-AC-9-16, CORRIDOR

Scenario	Fire Frequency	Damage Probability	CCDP	CDF (/yr)
Loss of Train B ECW due to 2/3L-378	1.81E-5	1.0	2.00E-4	3.63E-9
Loss of Train B ECW, no suppression	7.06E-3	2.0E-2	2.00E-4	2.82E-8
Automatic suppression available	7.06E-3	1.0	5.50E-6	3.80E-8
Transient fire with loss of Train B ECW	3.76E-4	3.8E-3	2.00E-4	2.86E-10
COMPARTMENT TOTAL				7.0 E-8

Results

The results of this evaluation indicate that fire compartment 2-AC-9-16, when analyzed for its four worst case fire scenarios, is not fire risk significant. The CDF values for the four fire scenarios were added to obtain a revised overall CDF value of 7.0E-8, thus allowing this fire compartment to screen. No further evaluation is necessary.

4.4.2.4 Fire Compartment 2-AC-9-17, Relay Room (Auxiliary Control Building, 9')

In order to better estimate the core damage frequency for this fire compartment, six separate fire scenarios were evaluated: a small fixed ignition source fire, four large fixed ignition source fires (each with five subscenarios for the major ignition sources), and a transient ignition source fire. The results for each of these scenarios are shown in Table 4.4-4. The initiating fire frequencies of the small fire scenario and large fire scenarios were based on a categorization of fires contained in the fire frequency database for PWR auxiliary building electrical cabinet fires, and combined with appropriate weighting factors as previously described. The auxiliary building electrical cabinet fires in the Fire Events Database [4-11] was divided into six (6) small fires (incident numbers 352, 361, 492, 666, 667, and 673) and nine (9) large fires (incident numbers 72, 74, 81, 90, 138, 177, 236, 285, and 399). Figure 4.4-3 provides a simplified compartment layout drawing.

TABLE 4.4-4
FIRE COMPARTMENT 2-AC-9-17, RELAY ROOM

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.44E-4	1.0	1.00E-4	1.44E-8
Large fire with loss of POS4 and POS17	sum of subscenarios below			1.15E-6
ignition source 2L-70	5.62E-5	1.0	1.2E-2	6.74E-7
ignition source 3L-70	5.62E-5	0.44	1.2E-2	2.96E-7
ignition source 2L-73	4.69E-5	0.173	1.2E-2	9.73E-8
ignition source 3L-73	4.69E-5	0.144	1.2E-2	8.10E-8
ignition source 2/3L-224	9.39E-6	0.008	1.2E-2	9.00E-10
Large fire with loss of POS4	similar to scenarios above			1.90E-9
Large fire with loss of POS17	similar to scenarios above			1.18E-9
Large fire with no significant losses	similar to scenarios above			5.34E-10
Transient fire with loss of POS4, POS17	3.76E-4	3.8E-3	1.2E-2	1.71E-8
COMPARTMENT TOTAL				1.2 E-6

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, with a resulting CDF of 1.44E-8.

The large fire scenarios consist of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. For the large fire scenarios, four major scenarios were developed to account for all of the potential damage states:

1. Loss of POS4 and loss of POS17

2. Loss of POS4

3. Loss of POS17

4. No significant losses (assumed loss of CVCS)

In addition, each of the above scenarios were subdivided by the potential fixed fire sources to account for the total fire ignition frequency for large fires and all potential combinations of significant component losses (POS4 and POS17) from all ignition sources in this fire compartment.

POS4 cables are located in cabinet 2L-70, as well as in various cable trays routed north to south above cabinet 2L-70. POS17 cables are located in cabinet 3L-70, as well as in several cable trays routed north to south through the center of this fire compartment (next to 3L-70). Fixed ignition sources in the room include electrical panels 2L-70, 3L-70, 2L-73, 3L-73, and 2/3L-224.

COMPBRN modeling was used to determine the probability of damaging POS4 and POS17 from each of the various ignition sources. POS4 raceways are represented by cable trays IFXU02 and IFXT02 for the purposes of modeling (see Figure 4.4-3). POS17 raceways are represented by cable trays IFXUP1 and IFXTP1 for the purposes of modeling. For each subscenario, COMPBRN fire modeling determined the probability of target damage and the mean time to damage for each of the offsite power targets. These values were then used to determine the probability of no manual suppression, modeled as an exponential function of the ratio of the time to damage versus the time to suppression [exp $(-t_d/t_s)$], based on NUREG/CR-2258, Fire Risk Analysis for Nuclear Power Plants [4-10].

Since there are no fire drill records and incidence reports for the 9' elevation of the Auxiliary Building, the records for the 50' elevation of the Auxiliary Building were used to calculate the mean time to manual suppression. These records are also used for the switchgear rooms on the 50' elevation. The mean time to manual suppression for the 50' elevation of the Auxiliary Building is assumed to be exponentially distributed with a mean time determined using fire department drill records and guidance provided in NUREG/CR-2258. Mean time to manual suppression for this fire compartment considered the following terms:

t_s = time for detection (2 min) + mean response time (9 min) + attack time (3 min) + time to mitigation (2 min)
= 16 minutes

The CDF for each of the subscenarios is therefore the product of the fire initiating frequency for the source, the probability of damage (which is calculated from the COMPBRN damage times and the manual suppression times), and the conditional CDP for that subscenario. The values calculated for each of the above subscenarios are added to determine the CDF for the large fire scenarios. The primary contributor to core damage frequency in this fire compartment is the loss of offsite power (POS4) to Unit 2 essential power in conjunction with the loss of offsite power (POS17) to Unit 3 essential power, where the dominant fire source is cabinet 2L-70.

The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 1.2E-6/yr. The fire compartment will be evaluated for containment heat removal and isolation capability (as discussed in Section 4.5) using the results of the IPE Level 2 Analysis.

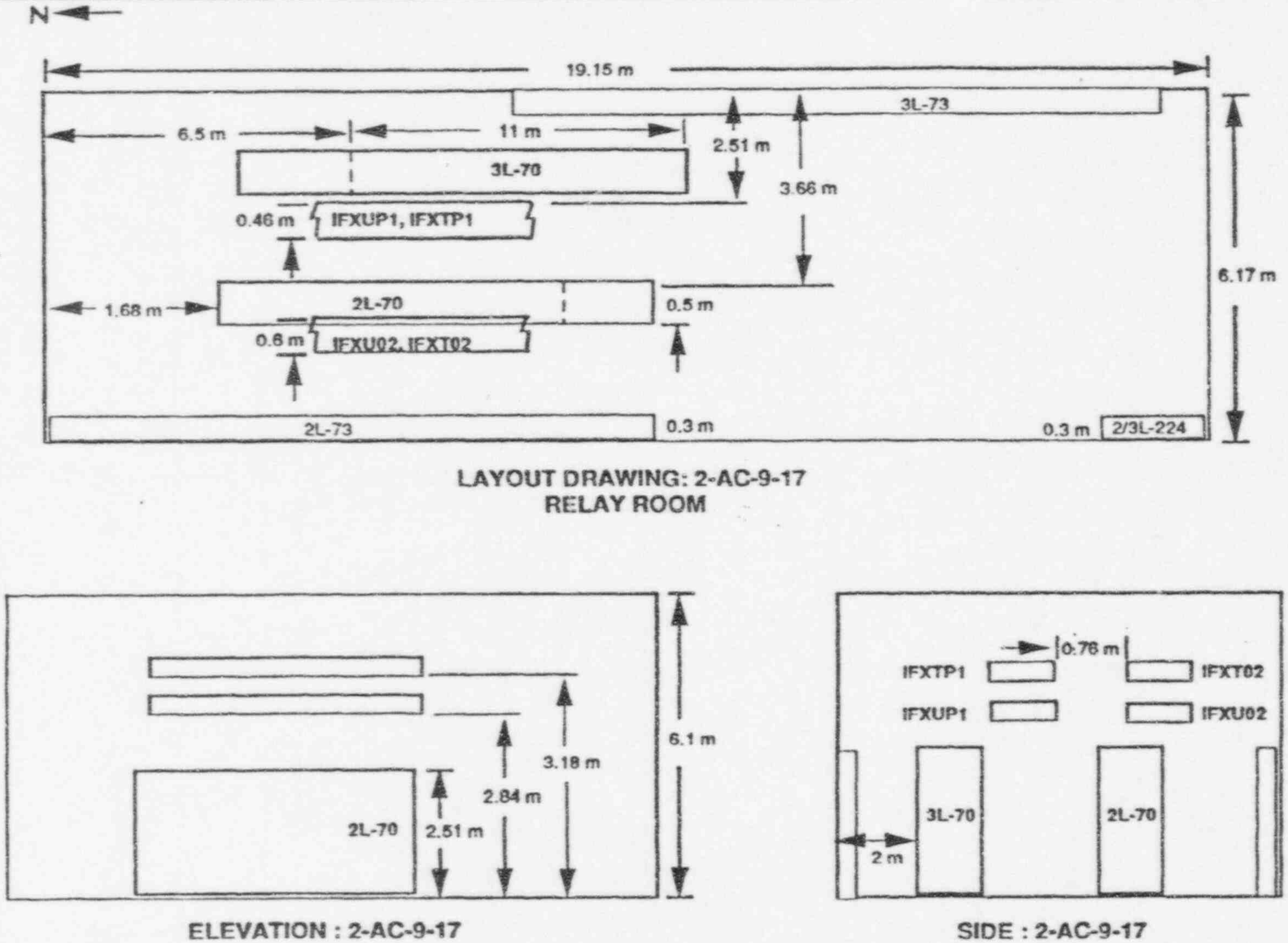


FIGURE 4.4-3

COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-AC-9-17, RELAY ROOM

4.4.2.5 Fire Compartment 2-AC-30-20A, Control Room (Auxiliary Control Building, 30')

Fire compartment 2-AC-30-20A is the common Unit 2 and 3 control room and cabinet areas, located on the 30' elevation of the Auxiliary Control Building. In order to better estimate the core damage frequency for this fire compartment, six separate fire scenarios were evaluated: five fixed ignition source fires (with subscenarios), and a transient ignition source fire. The fixed ignition source fire frequencies were revised based on a review of fires contained in the fire frequency database for control room fires. One fire (incident number 464) was determined to be not applicable to a fire in this compartment because it was a kitchen fire. The results for the control room area are presented in Table 4.4-5. Figures 4.4-4 and 4.4-5 provide simplified compartment models for the COMPBRN analyses.

TABLE 4.4-5
FIRE COMPARTMENT 2-AC-30-20A, CONTROL ROOM

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Fire in 2/3CR-63, loss of POS4/POS17	1.12E-4	1.0	2.9E-3	3.25E-7
Fire in 2CR-60 or 3CR-60	see subscenarios below			6.73E-8
loss of POS4/POS17	2.25E-4	0.1	2.9E-3	6.53E-8
no significant losses or evacuation	2.25E-4	0.9	1.0E-5	2.03E-9
Fire in 2L-34 and 2L-35	see subscenarios below			5.71E-8
loss of ESF auto-start	2.25E-4	3.4E-3	1.5E-3	1.15E-9
loss of AFW auto-start, with MFW	2.25E-4	0.997	2.5E-4	5.61E-8
Fire in 2CR-52 and adjacent panels	see subscenarios below			1.70E-7
loss of AFW auto-start and MFW	7.25E-4	0.1	1.4E-3	1.02E-7
loss of AFW auto-start, with MFW	2.25E-4	0.9	2.5E-4	5.06E-8
loss of MFW	5.00E-4	0.9	6.3E-5	2.84E-8
Fire in remaining ignition sources	see subscenarios below			1.72E-7
no significant losses, with evacuation	1.62E-2	3.4E-3	1.8E-4	9.91E-9
no significant losses, no evacuation	1.62E-2	0.997	1.0E-5	1.62E-7
Transient fire with evacuation	3.76E-4	3.8E-3	2.9E-3	4.14E-9
COMPARTMENT TOTAL				8.0E-7

The following fixed ignition source fire scenarios consider three primary targets based on their contributions to the control room CDF, the potential fire sources, and the success of manual suppression or the need for a control room evacuation:

1. Ignition source 2/3CR-63
   - Loss of POS4 and POS17, with control room evacuation

2. Ignition sources 2CR-60 and 3CR-60
   - Loss of POS4 and POS17, with control room evacuation
   - No significant component losses

3. Ignition sources 2L-34 and 2L-35
   - Loss of ESF auto-start capability
   - Loss of AFW auto-start capability, with MFW available

4. Ignition sources 2CR-52, and 2CR-50, 2CR-51, and 2CR-53
   - Loss of AFW auto-start capability, with loss of MFW
   - Loss of AFW auto-start capability, with MFW available
   - Loss of MFW

5. Remaining ignition sources
   - No significant losses, with evacuation
   - No significant losses, no evacuation

The primary contributors to core damage frequency in this fire compartment are the control room evacuation scenarios with loss of offsite power (POS4) to Unit 2 essential power and the loss of offsite power (POS17) to Unit 3 essential power. Offsite power cables for POS4 and POS17 are located in electrical control panel 2/3CR-63, adjacent to the west wall of the control room complex. Damage to this panel from the two adjacent panels (2CR-60 and 3CR-60) was assumed to occur in 10 minutes. A probability of failure to manually suppress the adjacent panel fires within 10 minutes was considered. COMPBRN modeling determined that damage to this panel from non-adjacent fixed ignition sources was insignificant. Figure 4.4-4 is a layout drawing depicting these fire scenarios. Control room evacuation was not a factor in this scenario because the CCDP for loss of POS4 and POS17 is the same with or without control room evacuation.

The secondary contributor to core damage frequency in this fire compartment is the loss of ESF auto-start capability. ESF cables are located in panels 2L-34 and 2L-35 in the northeast corner of the control room cabinet area. A probability of failure to manually suppress the fire within 20 minutes was included to account for control room evacuation. COMPBRN modeling determined that damage to this panel from other fixed ignition sources was insignificant. Figure 4.4-5 is a layout drawing depicting this fire scenario.

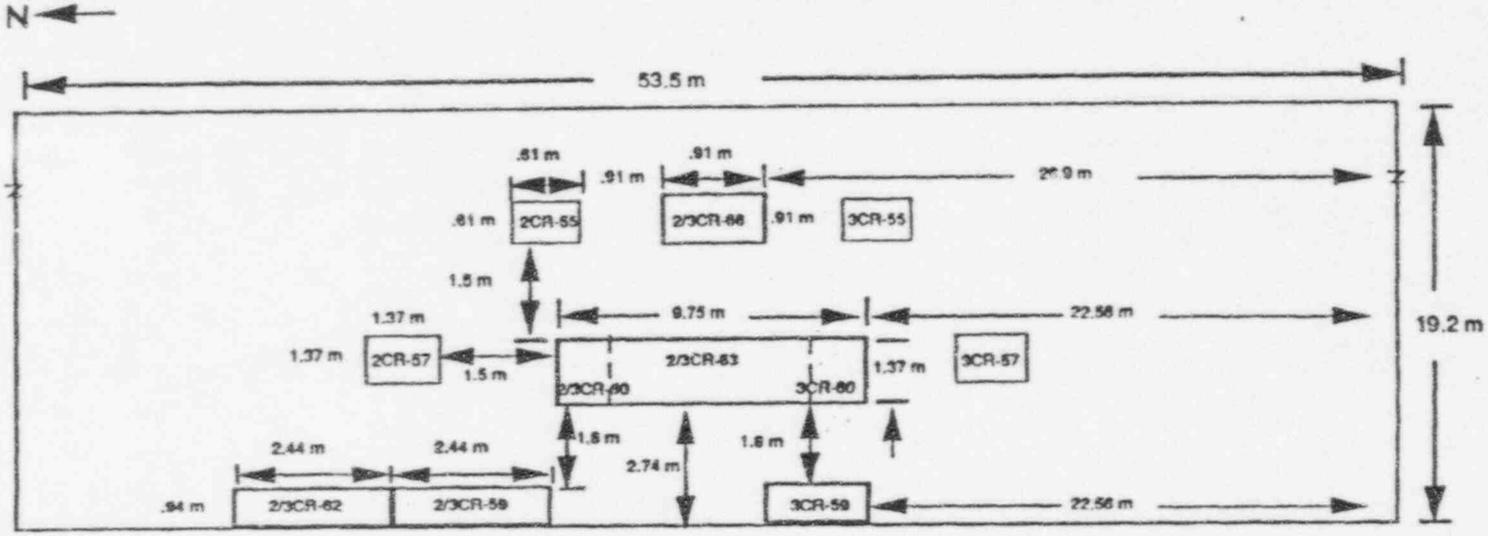
The final target considered in this fire compartment is the loss of auxiliary feedwater (AFW) auto-start capability, with or without MFW. Cables for AFW auto-start are located in panel 2CR-52. Cables for MFW auto-start are located in the adjacent panel, 2CR-53. Damage to 2CR-52 from the two adjacent panels (2CR-53 and 2CR-51) was assumed to occur in 10 minutes, and damage to 2CR-52 from 2CR-50 was assumed to occur in 20 minutes, based on guidance provided in Attachment H of the EPRI Fire PRA Implementation Guide. A probability of failure to manually suppress the adjacent panel fires within 10 minutes was included. COMPBRN modeling performed for panel 2/3CR-63 was used to determine that damage to 2CR-52 from other fixed ignition sources was insignificant.

The remaining fixed ignition sources which do not contribute to the above fire scenarios are assumed to cause a control room evacuation with no other significant component losses if manual suppression by control room personnel is not successful within 20 minutes. The probability of failing to extinguish the fire in that time is 3.4E-3 (NSAC 181). The probability of failure of manual suppression in 10 minutes for the above scenarios is assumed to be 0.1 (FIVE Methodology). This value can be supported by the Fire Events Database, in which 10 of the 11 control room fires were detected and extinguished by control room personnel (no information was provided for the remaining fire).

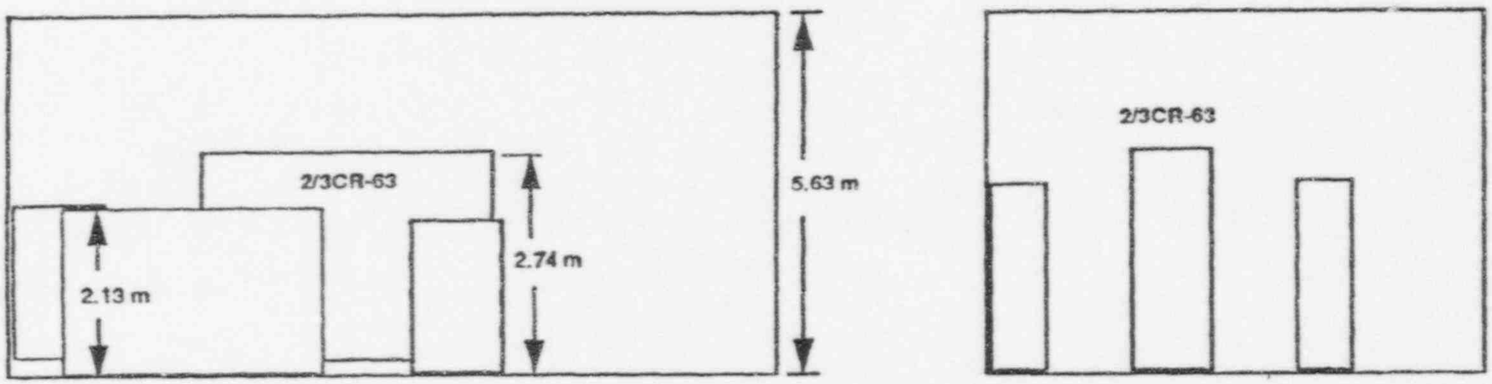
The transient fire scenario conservatively assumes that all transient fires in this compartment cause an initial loss of POS4 and POS17. The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for initial loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 8.0E-7/yr, which allows this compartment to screen from further analysis. While this CDF is close to the screening criterion of 1E-6/yr, there is significant conservatism in the analysis of the loss of offsite power scenarios, which comprise over 50% of this compartment CDF. Therefore, the screening of this compartment is conservative.



PARTIAL LAYOUT DRAWING:
2-AC-30-20A
CONTROL ROOM



ELEVATION : 2-AC-30-20A

SIDE : 2-AC-30-20A

FIGURE 4.4-4

TARGET: 2/3CR-63
COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-AC-30-20A, CONTROL ROOM

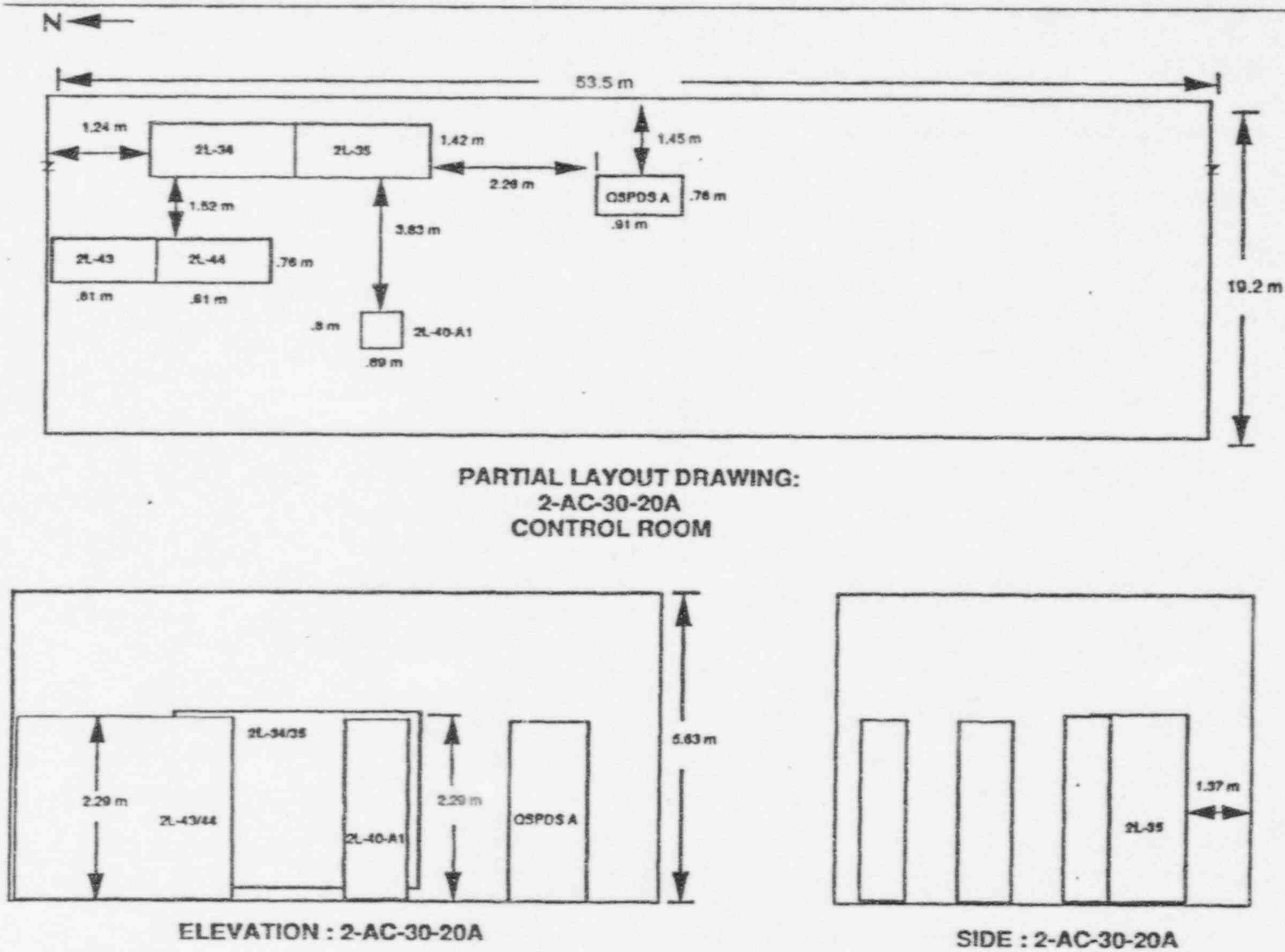


FIGURE 4.4-5
TARGETS: 2L-34 & 2L-35
COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-AC-30-20A, CONTROL ROOM

4.4.2.6 Fire Compartment 2-AC-30-20C, Computer Room (Auxiliary Control Building, 30')

Fire compartment 2-AC-30-20C is the computer room located on the 30' elevation of the Auxiliary Control Building adjacent to the control room complex. The Phase II evaluation was dominated by the assumption that a control room evacuation would occur if the computer room had a fire. Therefore, the probability of control room evacuation in the event of a fire in the computer room was reassessed by reviewing the evacuation requirements of Abnormal Operating Instruction S023-13-21, "Fire." This procedure recommends shutdown outside the control room for the scenarios listed below. Included is a discussion regarding the applicability of each recommendation to a fire in 2-AC-30-20C.

1. Significant erratic indications and/or loss of equipment control - The Appendix R safe shutdown analysis was reviewed in order to determine the extent of erratic indication which could be expected to occur due to the loss of safe shutdown equipment in this fire compartment, consisting primarily of electrical panels, HVAC dampers, and spurious RCS indications. In addition, any or all ESF functions for Unit 2 may spuriously actuate. However, per the UFHA, proceduralized de-energization and subsequent operation of safe shutdown equipment will mitigate any effects of undesired repositionings. No essential RCS process monitoring instrumentation is lost for a fire in this compartment. Therefore, significant erratic indication and/or loss of equipment control is not expected to occur.

2. Loss of control room HVAC to support control room equipment - A fire in this compartment may cause spurious closure of HVAC fire dampers. However, the control room train A essential HVAC unit and associated dampers remain available to support control room equipment.

3. Significant fire damage - The fire compartment boundaries in 2-AC-30-20C were evaluated per the UFHA. The fire boundaries and associated fire protection features were found to be adequate to prevent the propagation of fire beyond the compartment boundaries.

   Per the UFHA, the fixed temperature rate of rise heat detector and ionization detectors are expected to detect the fire within the first few minutes of the growth period of the fire and alert the control room for prompt action by the fire department. Portable extinguishers are available in the adjacent fire compartment, and a halon suppression system is also provided in the zone.

Based on the UFHA evaluation, signific .nt fire damage is not expected to occur for a fire in this compartment.

Results

Based on the above assessment, it is highly unlikely that control room evacuation will occur for a fire in this compartment. Elimina ing the need for control room evacuation results in a significantly reduced CCDP valu based on loss of POS4 (1E-4) and a revised overall CDF of 8.3E-8. Since the C F is <1E-6/yr (screening criteria), no further analysis is recommended.

4.4.2.7 Fire Compartment 2-AC-50-35 Switchgear Room 2B (Auxiliary Control Building, 50')

In order to better estimate the core damage frequency for this fire compartment, three separate fire scenarios were evaluated: a small fixed ignition source fire, a large fixed ignition source fire, and a transient ignition source fire. The results for each of these scenarios is shown in Table 4.4-6. The initiating fire frequencies of the small fire scenario and large fire scenario were based on a categorization of fires contained in the fire frequency database for switchgear fires, and combined with the appropriate weighting factors as previously described (refer to fire compartment 2-PE-45-3A).

TABLE 4.4-6
FIRE COMPARTMENT 2-AC-50-35, SWITCHGEAR ROOM 2B

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.00E-4	1.06E-7
Large fire with loss of POS4, Train B switchgear, and MFW	1.06E-3	1.0	2.80E-3	2.97E-6
Transient fire with loss of POS4, Train B switchgear, and MFW	3.14E-5	3.8E-3	2.80E-3	3.34E-10
COMPARTMENT TOTAL				3.1 E-6

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, because loss of a single breaker cabinet could not cause loss of Train B 4kV switchgear 2A06. Therefore, the fire frequency for small fires in this compartment was conservatively multiplied with the conditional CDP for loss of POS4, to obtain the small fire core damage frequency (CDF) of 1.06E-7/yr.

The large fire scenario consists of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. The primary fire impact in this fire compartment is the loss of Train B switchgear 2A06 resulting in the loss of train B essential power. Support components and associated cables for Train B essential power are routed extensively throughout the fire compartment, and it was conservatively assumed that any large fire would result in loss of Train B essential power. The secondary impacts in this fire compartment are the loss of offsite power (POS4) to Unit 2, and consequently, loss of Main Feedwater (MFW). Loss of POS4 would disable MFW due to loss of the 4KV switchgear which supplies power to the MFW system. Offsite power cables which could potentially trip POS4 are routed above 2A06 and 2B06 in this compartment. It is also assumed that a large fire would cause sufficient damage to Train B switchgear and cables to require the operators to procedurally trip offsite power (POS4) to mitigate the spurious operation of safety equipment. The panel associated with Main Feedwater (2L-435) and associated trays and conduits are also located in proximity to 2A06. COMPBRN modeling was not used in this scenario due to the proximity of the major contributors to core damage frequency in this fire compartment, as well as the procedural requirements to trip offsite power.

The transient fire scenario considers target damage to 2A06 and Main Feedwater (POS4). The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of 2A06, POS4, and MFW to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 3.1E-6/yr. The fire compartment will be evaluated for containment heat removal and isolation capability (as discussed in Section 4.5) using the results of the IPE Level 2 Analysis.

4.4.2.8 Fire Compartment 2-AC-50-40, Switchgear Room 2A (Auxiliary Control Building, 50')

In order to better estimate the core damage frequency for this fire compartment, three separate fire scenarios were evaluated: a small fixed ignition source fire, a large fixed ignition source fire, and a transient ignition source fire. The results for each of these scenarios is shown in Table 4.4-7. The initiating fire frequencies of the small fire scenario and large fire scenario were based on a categorization of fires contained in the fire frequency database for switchgear fires, and the use of the appropriate weighting factors as previously described (refer to fire compartment 2-PE-45-3A).

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, because loss of a single breaker cabinet could not cause loss of Train A 4kV switchgear 2A04. Therefore, the fire frequency for small fires in this compartment was conservatively multiplied with the conditional CDP for loss of POS4, to obtain the small fire core damage frequency (CDF) of 1.06E-7/yr.

TABLE 4.4-7
FIRE COMPARTMENT 2-AC-50-40, SWITCHGEAR ROOM 2A

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.00E-4	1.06E-7
Large fire with loss of POS4, Train A switchgear, and MFW	1.06E-[illegible]	1.0	3.10E-3	3.29E-6
Transient fire with loss of POS4, Train A switchgear, and MFW	3.14E-5	3.8E-3	3.10E-3	3.70E-10
COMPARTMENT TOTAL				3.4 E-6

The large fire scenario consists of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. The primary fire impact in this fire compartment is the loss of Train A switchgear 2A04 resulting in the loss of train A essential power. Support components and associated cables for Train A essential power are routed extensively throughout the fire compartment, and it was conservatively assumed that any large fire would result in loss of Train A essential power. The secondary impacts in this fire compartment are the loss of offsite power (POS4) to Unit 2, and consequently, loss of Main Feedwater (MFW). Loss of POS4 would disable MFW due to loss of the 4KV switchgear which supplies power to the MFW system. Offsite power cables which could potentially trip POS4 are routed above 2A04 in this compartment. It is also assumed that a large fire would cause sufficient damage to Train B switchgear and cables to require the operators to procedurally trip offsite power (POS4) to mitigate the spurious operation of safety equipment. The panels associated with Main Feedwater (2L-344 and 2L-396) and associated trays and conduits are also located in proximity to 2A04. COMPBRN modeling was not used in this scenario due to the proximity of the major contributors to core damage frequency in this fire compartment, as well as the procedural requirements to trip offsite power.

The transient fire scenario considers target damage to 2A04 and Main Feedwater (POS4). The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of 2A04, POS4, and MFW to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 3.4E-6/yr. The fire compartment will be evaluated for containment heat removal and isolation capability (as discussed in Section 4.5) using the results of the IPE Level 2 Analysis.

4.4.2.9 Fire Compartment 2-AC-85-70, Non-Essential Switchgear Room (Auxiliary Control Building, 85')

Fire compartment 2-AC-85-70 is the non-essential switchgear room located on the Unit 2 side of the 85' elevation of the Auxiliary Control Building. Six separate fire scenarios were evaluated: a small fixed ignition source fire, four large fire damage scenarios (each with five fixed ignition source subscenarios), and a transient ignition source fire. The results for each of these scenarios is shown in Table 4.4-8. The initiating fire frequencies of the small fire scenario and large fire scenario were based on a categorization of fires contained in the fire frequency database for switchgear fires, and the use of the appropriate weighting factors as previously described (refer to fire compartment 2-PE-45-3A).

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, with a resulting CDF of 1.06E-7.

TABLE 4.4-8
FIRE COMPARTMENT 2-AC-85-70, NON-ESSENTIAL SWITCHGEAR ROOM

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.00E-4	1.06E-7
Large fire with loss of POS4 and POS17	sum of subscenarios below			6.42E-7
ignition source 2A08	2.79E-4	0.091	1.1E-2	2.79E-7
ignition source 2B16	1.39E-4	0.091	1.1E-2	1.39E-7
ignition source 2B10	2.23E-4	0.091	1.1E-2	2.23E-7
ignition source 2A09	2.79E-4	0.0	1.1E-2	0.0
ignition source 2B15	1.39E-4	0.0	1.1E-2	0.0
Large fire with loss of POS4	similar to scenarios above			5.79E-8
Large fire with loss of POS17				0.0
Large fire with no significant losses				4.22E-9
Transient fire with loss of POS4, POS17	3.14E-5	3.8E-3	1.1E-2	1.31E-9
COMPARTMENT TOTAL				8.1 E-7

The large fire scenarios consist of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. For the large fire scenarios, four major scenarios were developed to account for all of the potential damage states:

1. Loss of POS4 and loss of POS17

2. Loss of POS4

3. Loss of POS17

4. No significant losses (assumed loss of CVCS)

In addition, each of the above scenarios were subdivided by the potential fixed fire sources to account for the total fire ignition frequency for large fires and all potential combinations of significant component losses (POS4 and POS17) from all ignition sources in this fire compartment.

POS4 cables are located in various cable trays leading to 4 kV non-essential switchgear 2A08 and 2A09, as well as in cable tray PRXBA1 located below PRXAA1.

POS17 cables are located in cable tray PRXAA1, which is located along the east wall of this fire compartment. Fixed ignition sources in the room include switchgear 2A08 and 2A09, as well as buses 2B15, 2B16, and 2B10. COMPBRN modeling was used to determine the probability of damaging POS4 and POS17 from the various ignition sources. Figure 4.4-6 is a layout drawing depicting the fire scenarios for this fire compartment.

The five subscenarios involve fires in 2A08, 2B16, 2B10, 2A09 and 2B15, respectively. Each of these ignition sources is considered a separate subscenario, involving a separate probability of damage and fire frequency, based on a fraction of the fire frequency for large fires. In subscenarios 2 and 5, COMPBRN modeling was used to determine the probability of damage to POS17 (PRXAA1) from 2B16 and 2B15. The probability of damage to POS17 from 2B10 (subscenario 3) was assumed to be equal to the value calculated for 2B16. COMPBRN modeling was also used to determine the probability of damage to POS4 (2A08, PRXCA2, PRXBA1) from 2B16. Damage to POS4 from 2B15 and 2B10 was determined from COMPBRN results for 2B16, as well as results for damage to PRXAA1 from 2B15, because of similar geometry between ignition sources and targets. In subscenarios 1 and 4 (fire in 2A08 and 2A09), POS4 is assumed to be lost. Due to similar geometry with regard to the target and fire source, damage to POS17 (PRXAA1) from a fire in 2A08 and 2A09 was assumed to be equivalent to the probability values determined for a fire in 2B16 and 2B15, respectively.

The fire source ignition frequency, damage probability, and CCDP were then multiplied together for each of the above subscenarios, and then added together to determine the CDF for the large fire scenario with loss of POS4 and POS17. Similarly, this process is performed for less damaging fire events from the same five fire sources. The large fire scenario CDFs are given in Table 4.4-8.

The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 8.1E-7/yr, thus allowing this fire compartment to be screened out. No further analysis is necessary.

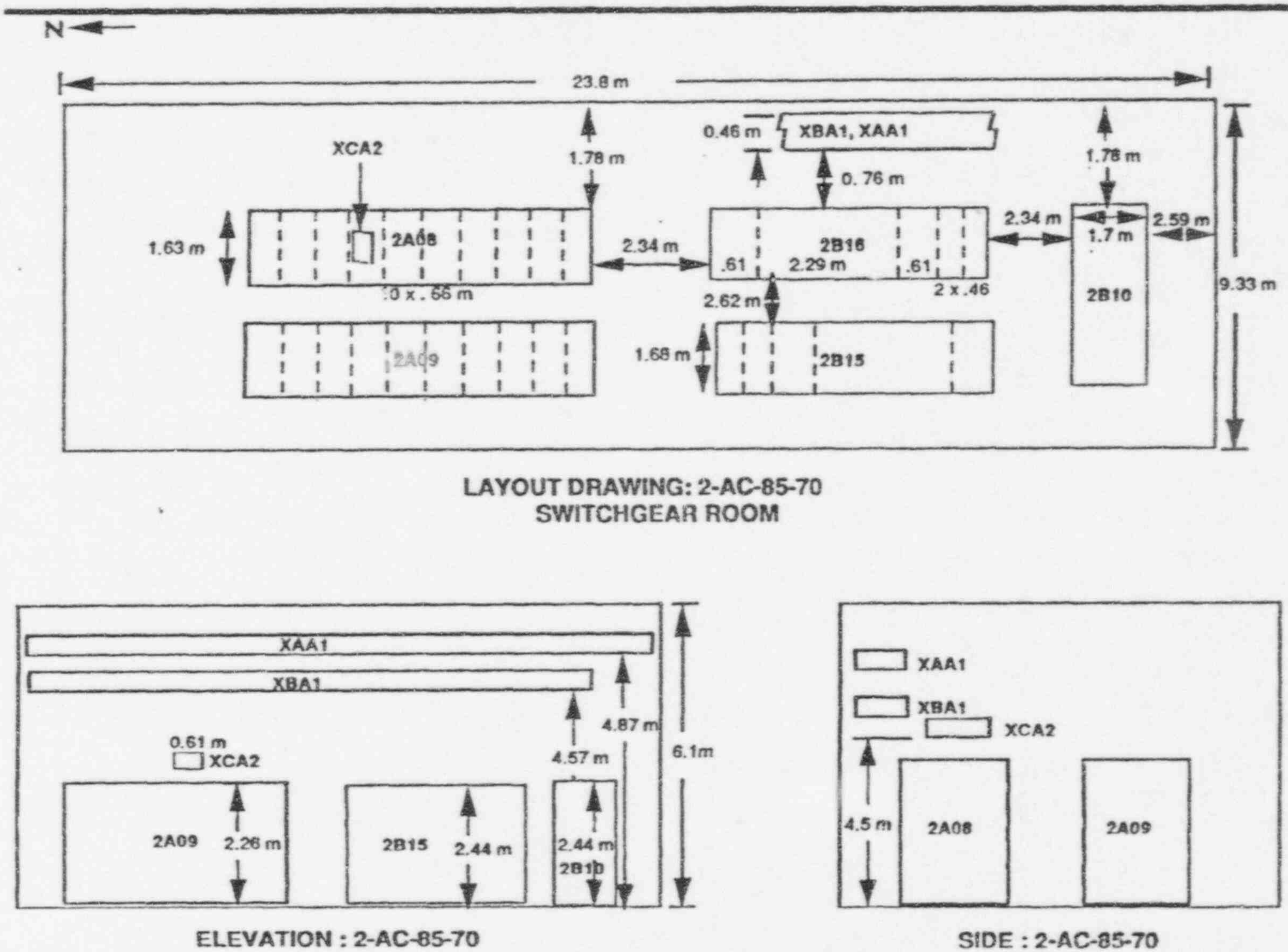


FIGURE 4.4-6
COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-AC-85-70

4.4.2.10 Fire Compartment 2-AC-85-71, Non-Essential Switchgear Room (Auxiliary Control Building, 85')

Fire compartment 2-AC-85-71 is the non-essential switchgear room located on the Unit 3 side of the 85' elevation of the Auxiliary Control Building, and the Phase III analysis is very similar to the previous fire compartment. However, the cabinets and cable trays are located closer together in this compartment, which results in a small increase in core damage frequency.

Six separate fire scenarios were evaluated: a small fixed ignition source fire, four large fire damage scenarios (each with four fixed ignition source subscenarios), and a transient ignition source fire. The results for each of these scenarios is shown in Table 4.4-9. The initiating fire frequencies of the small fire scenario and large fire scenario were based on a categorization of fires contained in the fire frequency database for switchgear fires, and the use of the appropriate weighting factors as previously described (refer to fire compartment 2-PE-45-3A).

TABLE 4.4-9
FIRE COMPARTMENT 2-AC-85-71, NON-ESSENTIAL SWITCHGEAR ROOM

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Small fire with loss of POS17	1.06E-3	1.0	2.5E-5	2.65E-8
Large fire with loss of POS4 and POS17	sum of subscenarios below			1.06E-6
ignition source 3A08	3.53E-4	0.182	1.1E-2	7.07E-7
ignition source 3B16	1.77E-4	0.182	1.1E-2	3.54E-7
ignition source 3A09	3.53E-4	0.0	1.1E-2	0.0
ignition source 3B15	1.77E-4	0.0	1.1E-2	0.0
Large fire with loss of POS4	similar to scenarios above			0.0
Large fire with loss of POS17	similar to scenarios above			2.07E-8
Large fire with no significant losses	similar to scenarios above			1.35E-9
Transient fire with loss of POS4, POS17	3.14E-5	3.8E-3	1.1E-2	1.31E-9
COMPARTMENT TOTAL				1.1 E-6

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS17) to Unit 3 essential power, because POS4 cables are not located in electrical cabinets in this compartment. Therefore, the fire frequency for small fires was conservatively multiplied with the conditional CDP for loss of POS17, to obtain the small fire CDF. Note that a

Unit 2 reactor trip must be assumed to obtain a CDF value for this scenario. The small fire scenario has a resulting CDF of 2.65E-8/yr.

The large fire scenarios consist of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. For the large fire scenarios, four major scenarios were developed to account for all of the potential damage states:

1. Loss of POS4 and loss of POS17

2. Loss of POS4

3. Loss of POS17

4. No significant losses (assumed loss of CVCS)

In addition, each of the above scenarios were subdivided by the potential fixed fire sources to account for the total fire ignition frequency for large fires and all potential combinations of significant component losses (POS4 and POS17) from all ignition sources in this fire compartment.

POS17 cables are located in various cable trays leading to 4 kV non-essential switchgear 3A08 and 3A09, as well as in cable tray PSXAA2 located above PSXBA2. POS4 cables are located in cable tray PSXBA2, which is routed along the east wall of this fire compartment. Fixed ignition sources in the room include switchgear 3A08 and 3A09, as well as buses 3B15 and 3B16. COMPBRN modeling was used to determine the probability of damaging POS4 and POS17 from the various ignition sources. Figure 4.4-7 is a layout drawing depicting the fire scenarios for this fire compartment.

The four subscenarios involve fires in 3A08, 3B16, 3A09 and 3B15, respectively. Each of these ignition sources is considered a separate subscenario, involving a separate probability of damage and fire frequency, based on a fraction of the fire frequency for large fires. In subscenarios 2 and 4, COMPBRN modeling was used to determine the probability of damage to POS4 (PSXBA2) from 3B16 and 3B15. COMPBRN modeling was also used to determine the probability of damage to POS17 (3A08, PSXCA2, PSXAA2) from 3B16. Damage to POS17 from 3B15 was determined from COMPBRN results for 3B16, as well as results for damage to PSXBA2 from 3B15, because of similar geometry between ignition sources and targets. In subscenarios 1 and 3 (fire in 3A08 and 3A09), POS17 is assumed to be lost. Due to similar geometry with regard to the target and fire source, damage to POS4 (PSXBA2) from a fire in 3A08 and 3A09 was assumed to be equivalent to the probability values determined for a fire in 3B16 and 3B15, respectively.

The fire source ignition frequency, damage probability, and CCDP were then multiplied together for each of the above subscenarios, and then added together to determine the CDF for the large fire scenario with loss of POS4 and POS17. Similarly, this process is performed for less damaging fire events from the same four fire sources. The large fire scenario CDFs are given in Table 4.4-9.

The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The value for the probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 1.1E-6/yr. The fire compartment will be evaluated for containment heat removal and isolation capability (as discussed in Section 4.5) using the results of the IPE Level 2 Analysis.

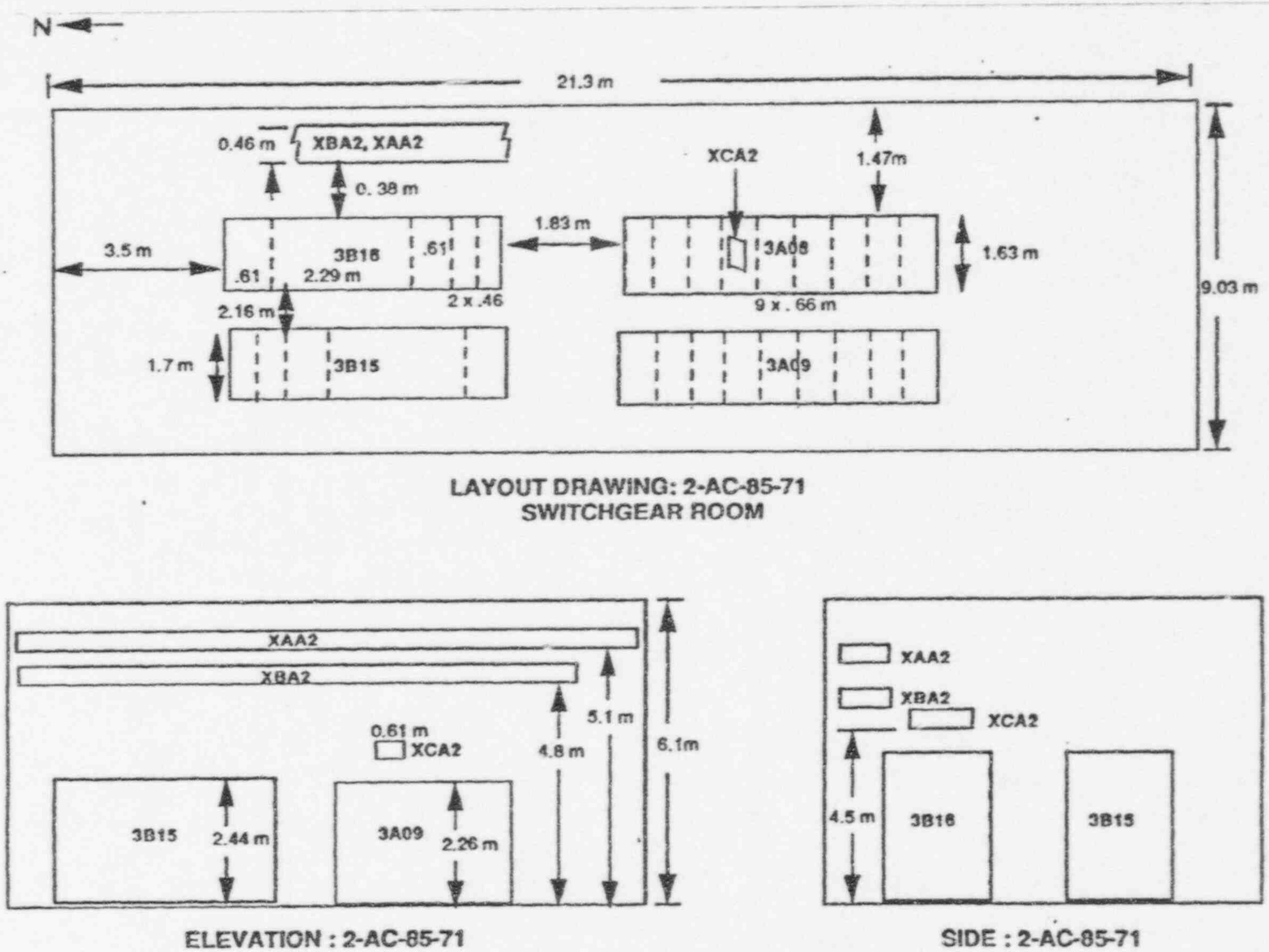


FIGURE 4.4-7

COMPBRN COMPUTER MODEL FOR FIRE COMPARTMENT 2-AC-85-71

4.4.2.11 Fire Compartment 2-AR-63-116, Corridor & Rooms (Auxiliary Radwaste Bldg, 63')

In order to better estimate the overall CDF value, fault tree modeling was performed on sets of fire scenarios for this compartment. Based on the location and configuration of offsite power relative to fixed ignition sources within the fire compartment, the following 3 fire scenarios were considered:

Scenario 1: Loss of offsite power to Unit 2 and Unit 3 essential power due to panel 2/3RT-7808.

Scenario 2: Loss of CVCS and MFW due to remaining fixed ignition sources.

Scenario 3: Loss of offsite power to Unit 2 and Unit 3 essential power due to transient fire sources.

In each case, revised fire frequency values and CCDP values were applied to the scenarios to better represent credible ignition sources for a given target and more accurate CCDP values based on specific target loss. The overall CDF values for the 3 fire scenarios were then added to obtain a revised overall CDF value. Table 4.4-10 presents the results for each of the three scenarios.

TABLE 4.4-10
FIRE COMPARTMENT 2-AR-63-116, RADWASTE CORRIDOR/ROOMS

Scenar	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Loss of POS4 and POS17 ue to 2/3RT-7808	1.82E-5	1.0	1.1E-2	2.00E-7
Loss of CVCS and MFW due to remaining fixed ignition sources	1.14E-3	1.0	7.0E-5	7.95E-8
Transient fire with loss of POS4, POS17	3.76E-4	3.8E-3	1.1E-2	1.57E-8
COMPARTMENT TOTAL				3.0 E-7

The first scenario considers loss of offsite power to Unit 2 (POS4) and Unit 3 (POS17) essential power due to a fire originating in panel 2/3RT-7808. This panel is the only ignition source located directly beneath the offsite power cable trays. The fire frequency for one electrical panel was applied to the CCDP for loss of POS4 and POS17 to obtain the CDF for this fire scenario.

The remaining fixed ignition sources in this fire compartment consist of pumps and electrical panels located in rooms separated from the corridor. Due to the size of this fire compartment, the heat release required to damage the offsite power cables in a hot gas layer is not achievable from the remaining fixed ignition sources. To obtain the CDF value for this scenario, the fire frequency for the remaining fixed ignition sources was applied to the CCDP value for loss of CVCS and MFW.

This scenario considers loss of offsite power to Unit 2 (POS4) and Unit 3 (POS17) essential power due to a transient fire. To determine the CDF for this scenario, the fire frequency for transient ignition sources was multiplied by the probability of transient combustible exposure (P_{tc}) and the CCDP for loss of POS4 and POS17.

Results

The results of this evaluation show that fire compartment 2-AR-63-116 is not fire risk significant. The CDF values for the three fire scenarios were added to obtain a revised overall CDF value of 3.0E-7 thus allowing this fire compartment to be screened. No further evaluation is necessary.

4.4.2.12 Fire Compartment 2-TB-148, Turbine Building

Fire compartment 2-TB-148 is located in the turbine building area and is comprised of the following SONGS 2/3 fire area/zones: 2-TB-7-148A (turbine building), 2-TB-7-148B (Unit 2/3 access road), 2-TB-9-148C (pump heat exchanger area), 2-TB-34-148D (Unit 2/3 access bridge), 2-TB-(-9)-148E (Unit 2/3 saltwater cooling pipe tunnel), 2-TB-9-148F (Unit 2 saltwater cooling pump room), 2-TB-8-148G (corridor), 2-TB-30-148H (FFCDP area), and 2-TB-72-154A (turbine generator and condenser).

In order to better estimate the overall CDF for this fire compartment, the following approach was taken. Based on the location and configuration of the primary contributors to CDF relative to fixed ignition sources within the fire compartment, one transient fire scenario and seven fixed fire scenarios below were considered. In each case, revised fire frequencies were applied to the scenarios to better represent credible ignition sources for a given target and more accurate CCDP were used based on specific target loss. The overall CDF values for the seven fire scenarios were then added to obtain a revised overall CDF value, which is presented in Table 4.4-11.

TABLE 4.4-11
FIRE COMPARTMENT 2-TB-148, TURBINE BUILDING

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Loss of POS4 and POS17 due to transient fire	2.29E-4	4.29E-2	1.2E-2	1.18E-7
Loss of POS4 due to fixed sources	1.47E-3	1.0	1.0E-4	1.47E-7
Loss of CCW due to loss of SWC pumps (auto suppression fails)	3.6 E-3	2.00E-2	3.0E-3	2.16E-7
Loss of 1 train of CCW (auto suppression successful)	3.6 E-3	0.98	3.0E-5	1.06E-7
Loss of MFW (water sprays unavailable)	2.25E-2	5.0 E-2	6.3E-5	7.09E-8
Loss of 1 train of MFW (local water sprays available)	2.25E-2	0.95	2.5E-5	5.34E-7
Loss of MFW due to remaining sources	1.73E-2		6.3E-5	1.09E-6
COMPARTMENT TOTAL				2.3 E-6

The following fire scenarios encompass the total fire frequency and significant safe shutdown losses in the turbine building (2-TB-148):

1. Loss of offsite power (POS4) to Unit 2 and 3 (POS17) due to transient sources
2. Loss of POS4
3. Loss of component cooling water (due to loss of all 4 SWC pumps, automatic suppression unavailable)
4. Loss of one train of component cooling water (automatic suppression available)
5. Loss of MFW and condensate due to fixed ignition sources with suppression systems unavailable
6. Loss of one train of main feedwater due to fixed ignition sources with suppression systems available
7. Loss of main feedwater and condensate due to remaining fixed ignition sources

Scenario 1: The highest transient fire contributor to CDF in this fire compartment is the simultaneous loss of offsite power from Unit 2 and Unit 3 sources, which also fails MFW. A review of cable routing for POS4 and POS17 determined that Unit 3 offsite power (POS17) has limited exposure in this fire compartment. The only fire scenario which could cause the simultaneous loss of POS4 and POS17 is a fire in the access road, elevation 7', adjacent to the Unit 2 saltwater cooling pump room. In this area, POS17 cables are routed in cable tray JEXTA1 beside POS4 cables in cable tray JEXRA1. There are no fixed ignition sources below the cable trays. A transient fire was assumed to damage both cable trays. Conservatively, the transient fire frequency for the entire turbine building was included in this scenario as it has the highest CCDP. This frequency was multiplied by the transient combustible exposure (P_{tc}) value and the CCDP due to loss of POS4 and POS17. The resulting estimated core damage frequency is 1.18E-7/yr.

Scenario 2: POS4 cables are routed along the east wall of this fire compartment. The fire frequency was revised based on only the fixed ignition sources located in the vicinity of the routing. Due to the size of this fire compartment, the heat release required to damage the offsite power cables in a hot gas layer is not achievable from other fixed ignition sources. Included in the revised fire frequency are 8 pumps and 6 electrical panels. The CCDP was also revised to reflect loss of POS4. The routing of main feedwater (MFW) cables may follow a route similar to that of offsite power. However, loss of POS4 implies a loss of MFW due to loss of the 4KV switchgear that provides power to the MFW system. The CDF due to this fire scenario is 1.47E-7/yr.

Scenario 3: This scenario involves the loss of component cooling water due to loss of all SWC pumps combined with the probability of automatic suppression system failure. Based on a review of the cable routing for the four saltwater cooling pumps, it was determined that a loss of both trains of component cooling water could only occur in fire zone 2-TB-9-148F, the saltwater cooling pump room. SWC pumps 2P-113B and 2P-112A are located in this fire compartment, as well as cables for 2P-307A. In addition, cable 2BA06110G impacting pump 2P-114B is unwrapped in this fire compartment. Train B SWC pump cables located in fire zone 2-TB-(-9)-148E (Unit 2/3 saltwater cooling pipe tunnel) are provided with fire protective coating. Other SWC pump cables located in fire zone 2-TB-7-148B (Unit 2/3 access road) are indication cables only and do not adversely impact pump operation. The fire frequency was revised to consider a fire in the saltwater cooling pump room and was based on all pumps and electrical panels located in the room. The CCDP value was also revised to reflect the loss of both trains of saltwater cooling, leading to the loss of component cooling water. These values were multiplied by the unavailability of automatic suppression from wet pipe sprinklers in the pump room to obtain the CDF of 2.16E-7/yr for this scenario.

Scenario 4: This scenario involves the loss of one train of component cooling water, combined with the probability that the automatic suppression system is available. The fire frequency was revised to consider a fire in the saltwater cooling pump room and was based on all pumps and electrical panels located in the room. The CCDP value was also revised to reflect the loss of one train of saltwater cooling leading to the loss of one train of component cooling water. These values were multiplied by one minus the unavailability of automatic suppression from wet pipe sprinklers in the pump room. The CDF for this fire scenario is 1.06E-7/yr.

Scenario 5: Main feedwater equipment and cables are located extensively throughout the turbine building. This scenario considers the fire ignition sources in this fire compartment not covered in scenarios 1 through 4, which are protected by local suppression systems. The fire frequency value was revised to consider the contribution of the turbine generator oil and hydrogen seal oil as well as the main feedwater pumps. The CCDP, which was revised to reflect the loss of main feedwater, was multiplied by the unavailability of water spray systems. The CDF for this fire scenario is 7.09E-8/yr.

Scenario 6: This scenario has the same initiators as the previous scenario, but accounts for the probability that automatic suppression remains available. The fire frequency was revised to consider the contribution of the turbine generator oil and hydrogen seal oil as well as the main feedwater pumps. The CCDP value was revised to reflect with the loss of one train of main feedwater. These values were multiplied by one minus the unavailability of water spray systems. The CDF for this fire scenario is 5.34E-7/yr.

Scenario 7: This scenario considers the fire ignition sources in this fire compartment not covered in Scenarios 1 through 6. The revised fire frequency for this scenario is the total turbine building fire frequency, minus the combined fire frequencies of Scenarios 1-6. The CCDP was revised to reflect the loss of main feedwater. The CDF for this fire scenario is 1.09E-6/yr.

Results

The Phase III CDF values for the seven fire scenarios were added to obtain a revised overall CDF value of 2.3E-6/yr, and the fire compartment will be evaluated for containment heat removal and isolation capability as discussed in Section 4.5 using the results of the IPE Level 2 Analysis.

4.4.2.13 Fire Compartment 2-DG-30-155, Train B Diesel Generator Room

In order to better estimate the core damage frequency for this fire compartment, three separate fire scenarios were evaluated: a small fixed ignition source fire, a large fixed ignition source fire, and a transient ignition source fire. A small fire scenario and a large fire scenario were postulated based on a categorization of fires contained in the fire frequency database for diesel generator fires. Approximately 70% of the diesel generator fires surveyed were considered small fires and the remaining 30% were considered large fires (even though most of the 30% only involved the diesel, many had the potential for further room involvement). The resulting CDF for these scenarios is presented in Table 4.4-12.

TABLE 4.4-12
FIRE COMPARTMENT 2-DG-30-155, TRAIN B DIESEL GENERATOR ROOM

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Small fire with loss of train B DG	1.8 E-2	1.0	2.6E-5	4.73E-7
Large fire with loss of train B DG and initial loss of switchgear 2A06	1.0 E-2	1.0	7.5E-5	7.5 E-7
Transient fire with loss of train B DG and initial loss of switchgear 2A06	1.67E-4	3.8E-3	7.5E-5	4.76E-11
COMPARTMENT TOTAL				1.2 E-6

The small fire scenario consists of fire events which did not involve the room beyond the diesel itself. These fires typically were contained in the diesel engine or exhaust manifold/piping and did not damage any other equipment beyond the diesel. The contribution to core damage frequency in this scenario consists of loss of the Unit 2 train B diesel generator.

The large fire scenario consists of fire events which had a potential for room involvement beyond the diesel. The primary contributor to core damage frequency in this fire scenario is the loss of power to the train B switchgear, 2A06, due to fire-induced damage of the diesel generator supply breaker circuitry. Damage to this circuitry could potentially close the diesel breaker which would trip the offsite power supply breaker. These cables are located in the diesel generator control cabinet, 2L-161-B and in conduit routed behind the cabinet along the wall. The primary ignition source in this room is the diesel itself, however, potential ignition of electrical cabinets and plant-wide ignition sources is also included in this scenario. Upon loss of power to 2A06, operator action to recover the switchgear by disconnecting power to the diesel generator breaker and reclosing the offsite power breaker is credited. A procedural modification will be recommended to perform this action.

The transient fire scenario considers target damage to the train B diesel and the diesel control cabinet, 2L-161-B, due to a transient ignition source fire. The value for the probability of transient combustible exposure (P_{tc}) was calculated for fire compartment 2-DG-30-155 and applied to the fire frequency for transients and the CCDP value for loss of the train B diesel with operator action to recover 2A06. The CDF for this transient fire scenario is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 1.2E-6/yr. This fire compartment will be evaluated for containment heat removal and isolation capability as discussed in Section 4.5 using the results of the IPE Level 2 Analysis. A procedure change request to procedure SO23-13-21, "FIRE", will be initiated for implementation by the end of the Cycle 9 refueling outage. This is to ensure power recovery to switchgear 2A06 in the event of a fire in this compartment.

4.4.2.14 Fire Compartment 2-DG-30-158, Train A Diesel Generator Room

This fire compartment was analyzed similarly to the previous compartment for the train B DG, with the same resulting CDF. Three separate fire scenarios were evaluated: a small fixed ignition source fire, a large fixed ignition source fire, and a transient ignition source fire. A small fire scenario and a large fire scenario were postulated based on a categorization of fires contained in the fire frequency database for diesel generator fires. Approximately 70% of the diesel generator fires surveyed were considered small fires and the remaining 30% were considered large fires (even though most of the 30% only involved the DG, many had the potential for further room involvement). The resulting CDF for these scenarios is presented in Table 4.4-13.

TABLE 4.4-13
FIRE COMPARTMENT 2-DG-30-158, TRAIN A DIESEL GENERATOR ROOM

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Small fire with loss of train A DG	1.8 E-2	1.0	2.6E-5	4.73E-7
Large fire with loss of train A DG and initial loss of switchgear 2A04	1.0 E-2	1.0	7.5E-5	7.5 E-7
Transient fire with loss of train A DG and initial loss of switchgear 2A04	1.67E-4	3.8E-3	7.5E-5	4.76E-11
COMPARTMENT TOTAL				1.2 E-6

The small fire scenario consists of fire events which did not involve the room beyond the DG itself. These fires typically were contained in the diesel engine or exhaust manifold/piping and did not damage any other equipment beyond the DG. The contribution to core damage frequency in this scenario consists of loss of the Unit 2 train A diesel generator.

The large fire scenario consists of fire events which had a potential for room involvement beyond the DG. The primary contributor to core damage frequency in this fire scenario is the loss of power to the train A switchgear, 2A04, due to fire-induced damage of the diesel generator supply breaker circuitry. Damage to this circuitry could potentially close the diesel breaker which would trip the offsite power supply breaker. These cables are located in the diesel generator control cabinet, 2L-160-A and in conduit routed behind the cabinet along the wall. The primary ignition source in this room is the diesel generator itself. However, potential ignition of electrical cabinets and plant-wide ignition sources is also included in this scenario. Upon loss of power to 2A04, operator action to recover the switchgear by disconnecting power to the diesel generator breaker and reclosing the offsite power breaker is credited. A procedural modification to address this recommended operator action will be implemented by the end of Cycle 9 Outage.

The scenario considers target damage to the train A diesel generator and the diesel generator control cabinet, 2L-160-A, a transient ignition source fire. The value for the probability of transient combustible exposure (P_{tc}) was calculated for fire compartment 2-DG-30-158 and applied to the fire frequency for transients and the CCDP value for loss of the train A diesel with operator action to recover 2A04. The CDF for this transient fire scenario is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 1.2E-6/yr. As discussed in Section 4.5, this fire compartment will be evaluated for containment heat removal and isolation capability using the results of the IPE Level 2 Analysis. A procedure change request to procedure SO23-13-21, "FIRE", has been initiated for implementation by the end of the Cycle 9 refueling outage. This is to ensure power recovery to switchgear 2A04 in the event of a fire in this compartment.

4.4.2.15 Fire Compartment 2-YD-30-200, Unit 2 and 3 Yard Area

Fire compartment 2-YD-30-200 is the Unit 2/3 yard area, comprised of SONGS 2/3 fire area/zones 2-YD-30-200A (Unit 2) and 2-YD-30-200B (Unit 3). In order to better estimate the overall CDF for this fire compartment, the following approach was taken. Based on the location and configuration of the primary contributors to CDF relative to fixed ignition sources in the fire compartment area, the three fire scenarios below were considered. In each case, revised fire frequencies were applied to the scenarios to better represent credible ignition sources for a given target and more accurate CCDP were used based on specific target loss. The overall CDF for the three fire scenarios were then added to obtain a revised overall CDF, as presented in Table 4.4-14.

TABLE 4.4-14
FIRE COMPARTMENT 2-YD-30-200, UNIT 2 AND 3 YARD AREA

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/YR)
Loss of POS4 (fixed ignition sources)	1.6 E-3	1.0	1.0E-4	1.60E-7
Loss of POS4 and POS17 (transient)	2.02E-4	4.59E-2	1.2E-2	1.11E-7
Loss of POS17 (fixed ignition sources)	1.47E-2	1.0	2.5E-5	3.67E-7
COMPARTMENT TOTAL				6.4 E-7

Before each scenario was evaluated, the configuration of ignition sources in the yard area was reviewed to ensure adequate separation between scenarios. Per the UFHA, the largest amounts of combustibles in the yard area are concentrated in the transformers and lube oil tanks. The safe shutdown cabling of concern in this area is limited to the underground duct banks and manholes. Walkdowns were performed in order to determine the separation of redundant safe shutdown cables and equipment in the yard area. Relevant information is as follows:

Fire Scenarios in the Yard	Minimum Separation Distance
1E 4kV switchgear control cables, train A versus train B	83 feet
Offsite power POS4, POS17 cables versus train A and B essential power	273 feet

The closest A and B train raceways are AKA229 and AKB225, which are routed through manholes separated by approximately 50 feet, and are located within buildings B-64 and B-62, respectively. Since the manholes are located within separate, noncombustible buildings, a flammable liquids spill will not affect both manholes simultaneously. There is no continuous path of combustibles between the two buildings (other than asphalt pavement), and the ground is contoured to prevent the flow of liquids to both buildings. In addition, redundant trains of AFW are routed underground between the cable tunnel (2-142B) and Containment. There are no openings to the yard area from these trenches and the conduits in the Train B trench are buried in sand The review of fire scenario separation ensured that more than one of the postulated fire scenarios could not occur simultaneously.

The first scenario involves loss of POS4 due to loss of one reserve auxiliary transformer (2XR1, 2XR2 or 2XR3). Reserve auxiliary transformers 2XR1, 2XR2 and 2XR3 are located east of the Tank Building in the yard area. A fire was postulated to occur which would damage one of the transformers and result in a loss of offsite power to Unit 2 essential power. The fire frequency for this scenario was estimated based on a fire in any of the three offsite power transformers. The revised CCDP value reflects loss of POS 4 Unit 2 essential power. The CDF value of 1.6E-7/yr represents the risk contribution due to this fire scenario.

The second scenario involves loss of offsite power, POS4 and POS17, due to transient fire sources. Cables associated with Unit 2 and Unit 3 offsite power are routed in a conduit trench which extends underground from the east end of the Auxiliary Control Building east beyond the Radwaste Building. This conduit trench contains no fixed ignition sources. Similar conduit trenches include cables associated with train A essential power routed in a conduit trench which extends from the east end of the cable tunnel towards the Diesel Generator Building. Cables associated with train B essential power are routed in a conduit trench which extends underground from the west end of the cable tunnel towards the Diesel Generator Building. These trenches do not communicate with each other and contain no fixed ignition sources. Therefore, the fire frequency for transient fires was applied to the worst case fire scenario, loss of POS4 and POS17, as reflected in the revised CCDP value. A P_{tc} value was applied to quantify the probability of transient exposure in the yard area. The CDF for this transient fire scenario is 1.11E-7/yr.

The third scenario considers the remaining fire ignition sources in this fire compartment not covered in scenarios 1 and 2. The CCDP was revised to reflect fire-induced loss of POS17 (offsite power to Unit 3) with no Unit 2 reactor trip. The CDF for this fire scenario is 3.67E-7/yr.

Results

The results of this evaluation indicate that fire compartment 2-YD-30-200, when analyzed for its three fire scenarios, is not fire risk significant. The CDF for the three fire scenarios were added to obtain a revised overall CDF of 6.4E-7/yr, thus allowing this fire compartment to be screened. No further evaluations are necessary.

4.4.2.16 Fire Compartment 3-PE-45-3A, Electrical Penetration Room (Unit 3 Penetration Building, 45')

This compartment was analyzed similar to its Unit 2 counterpart, 2-PE-45-3A. Four separate fire scenarios were evaluated: a small fixed ignition source fire, two large fixed ignition source fires, and a transient ignition source fire. A small fire scenario and a large fire scenario were postulated based on a categorization of fires contained in the fire frequency database for switchgear fires (refer to fire compartment 2-PE-45-3A). This fire compartment was evaluated as a switchgear room because it contains 6.9 kV switchgear and associated cables, 480 V transformers and cabinets. The results for the scenarios are given in Table 4.4-15.

TABLE 4.4-15
FIRE COMPARTMENT 3-PE-45-3A, ELECTRICAL PENETRATION ROOM

Scenario	Fire Frequency	Probability of Damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.00E-4	1.06E-7
Large fire with loss of POS4and POS17	2.94E-4	1.0	2.9 E-3	8.53E-7
Large fire with loss of POS17	7.74E-4	1.0	2.5 E-5	1.94E-8
Transient fire with loss of POS4, POS17	3.14E-5	3.8E-3	1.1 E-2	1.31E-9
COMPARTMENT TOTAL				9.8 E-7

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, because loss of a single breaker cabinet could not cause loss of both POS4 and POS17 (offsite power to Unit 3 essential power). Therefore, the fire frequency for small fires in this compartment was conservatively multiplied with the conditional CDP for loss of POS4 to obtain the small fire core damage frequency of 1.06E-7/yr.

The large fire scenarios consist of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. Two separate large fire scenarios are evaluated, (1) loss of POS4 and POS17 due to a fire in switchgear 3A01, and (2) procedural loss of POS17 due to a fire in the remaining fixed ignition sources. All of these scenarios assume a procedural (SO23-13-21, "Fire") trip of POS17 to mitigate spurious actuations, due to the inaccessibility of 3A01 in the presence of a large fire in this compartment. The results of these scenarios are added to obtain the total CDF for large fires.

The primary contributor to core damage frequency in this fire compartment is the loss of offsite power (POS4) to Unit 2 essential power in conjunction with the loss of offsite power (POS17) to Unit 3 essential power. COMPBRN modeling determined that damage to both POS4 and POS17 could occur only from switchgear 3A01 (see 2-PE-45-3A). Based on the fire frequency of 3A01, and the CCDP associated with losses of POS4 and POS17, the scenario CDF was 8.53E-7/yr.

The remaining fixed ignition sources in this compartment are conservatively assumed to cause damage to all safe shutdown cables in the compartment, except POS4. This results in component losses (which include various Unit 3 CCW, CVCS and RCS components), and a procedural trip of POS17. This assumes the fire is large enough to prevent access to switchgear 3A01. The resulting CCDP was multiplied by the fire frequency of the remaining fixed ignition sources, including electrical panels and HVAC units, to obtain a CDF for this scenario of 1.94E-8/yr.

The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 9.8E-7/yr. While this compartment CDF is fairly close to the approved FIVE screening criterion of 1E-6/yr, all of the dominant scenarios involve loss of offsite power to both units, and random (non-fire related) failures of the diesel generators and diesel support systems. Because the internal events model used for this screening evaluation was very conservative in the treatment of diesel generator unavailability, the screening results are considered to be conservative and bounding. The fire compartment can be screened from further analysis.

4.4.2.17 Fire Compartment 3-PE-63-3B, Electrical Penetration Room (Unit 3 Penetration Building, 63')

This compartment evaluation is similar to the previous compartment, 3-PE-45-3A, and its Unit 2 counterpart, 2-PE-63-3B. Five separate fire scenarios were evaluated: a small fixed ignition source fire, three large fixed ignition source fires, and a transient ignition source fire. The results for each of these scenarios are shown in Table 4.4-16. The initiating fire frequencies of the small fire scenario and large fire scenarios were based on a categorization of fires contained in the fire frequency database for switchgear fires, and use of the appropriate weighting factors as previously described. This fire compartment was evaluated as a switchgear room because it contains 6.9 kV switchgear and associated cables, 480 V transformers and cabinets. Figure 4.4-8 is a layout drawing depicting this compartment.

TABLE 4.4-16
FIRE COMPARTMENT 3-PE-63-3B, ELECTRICAL PENETRATION ROOM

Scenario	Fire Frequency	Probability of damage	CCDP	CDF (/yr)
Small fire with loss of POS4	1.06E-3	1.0	1.0E-4	1.06E-7
Large fire with loss of POS4 and POS17 (3A02 and 3B08 sources)	6.48E-4	1.0	2.9E-3	1.88E-6
Large fire with loss of POS17 and POS4 (other fixed sources)	4.12E-4	0.31	2.9E-3	3.7 E-7
Large fire with loss of POS17	2.88E-4	1.0	2.5E-5	7.2 E-9
Transient fire with loss of POS4, POS17	3.14E-5	3.8E-3	1.1E-2	1.31E-9
COMPARTMENT TOTAL				2.3 E-6

The small fire scenario consists of fire events which are 5 minutes or less in duration and are confined to the originating cabinet, with very limited heat and/or smoke release. The worst case loss for this scenario is loss of offsite power (POS4) to Unit 2 essential power, because loss of a single breaker cabinet could not cause loss of both POS4 and POS17 (offsite power to Unit 3 essential power). Therefore, the fire frequency for small fires in this compartment was conservatively multiplied with the conditional CDP for loss of POS4 to obtain the small fire core damage frequency of 1.06E-7/yr.

The large fire scenarios consisting of fire events which are greater than 5 minutes in duration, involve more than one electrical cabinet, and generate enough heat and smoke to damage cable trays in the fire compartment. Three separate large fire scenarios are evaluated:

(1) loss of POS4 and POS17 due to a fire in switchgear 3A02 or bus 3B08,

(2) loss of POS4 and POS17 due to a fire in the remaining fixed ignition sources, and

(3) procedural loss of POS17 due to a fire in the remaining fixed ignition sources.

All of these scenarios assume a procedural (SO23-13-21, "Fire") trip of POS17 to mitigate spurious actuations due to the inaccessibility of 3A02 in the presence of a large fire in this compartment. The results of these scenarios are added to obtain the total CDF for large fires.

The primary contributor to core damage frequency in this fire compartment is the loss of offsite power (POS4) to Unit 2 essential power in conjunction with the loss of offsite power (POS17) to Unit 3 essential power. POS4 and POS17 cables are routed in several raceways from the west end of the compartment to the east end. 3A02 and 3B08 are located in the east end of the compartment, with offsite power raceways routed directly over bus 3B08 and entering into 3A02. COMPBRN modeling determined that damage to both POS4 and POS17 could occur from a large fire in switchgear 3A02 or from a fire damaging cable raceways above bus 3B08. Based on the fire frequency of 3A02 and 3B08, and the CCDP associated with losses of POS4 and POS17, the scenario CDF is 1.88E-6/yr.

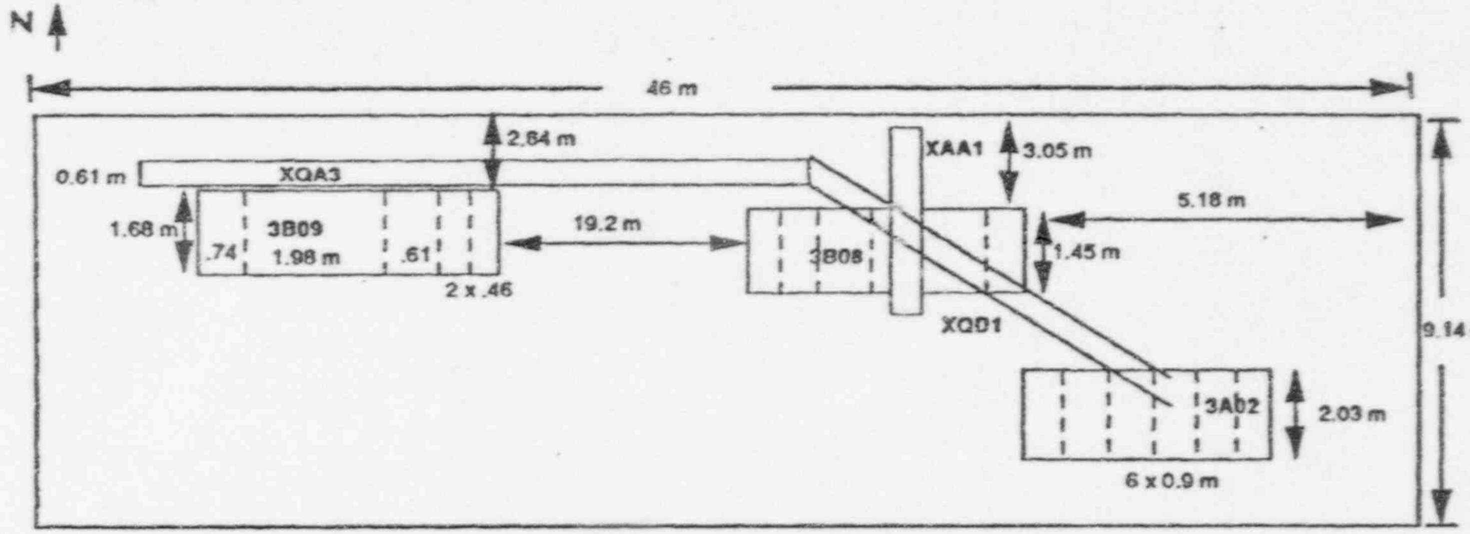
The secondary contributor to core damage frequency in this compartment is the loss of POS4 and POS17 due to a fire in the remaining fixed sources. Using COMPBRN modeling from the Unit 2 similar compartment for the worst case fire source and raceway, fire was determined to spread to POS4 and POS17 cables with a probability of 0.31. Loss of POS17 could also be caused by a procedural trip. The CDF of this scenario is 3.7E-7/yr.

Except those evaluated in the previous scenarios, the remaining fixed ignition sources in this compartment are conservatively assumed to cause damage to all safe shutdown cables in the compartment. This results in component losses which include various CCW, CVCS and RCS components and a procedural trip of POS17. This assumes the fire is large enough to prevent access to switchgear 3A02. The resulting CCDP was multiplied by the fraction of the fire frequency of the remaining fixed ignition sources not used in the previous scenario, including electrical panels and an HVAC unit, to obtain a CDF for this scenario of 7.2E-9/yr.

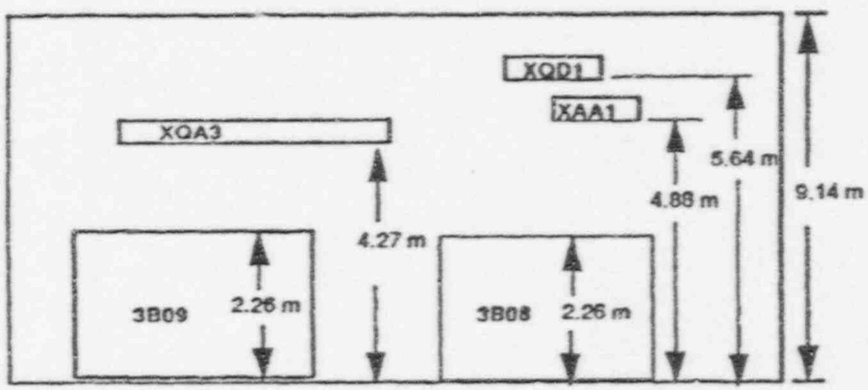
The transient fire scenario conservatively assumes that all transient fires in this compartment cause a loss of POS4 and POS17. The probability of transient combustible exposure (P_{tc}), which includes consideration of the type and quantity of transient combustibles as well as inspection frequencies, was calculated using the FIVE methodology. This value, 3.8E-3, was applied to the fire frequency for transients and the CCDP value for loss of POS4 and POS17 to obtain the CDF for this fire scenario. As with all transient fire scenarios at SONGS, the CDF is not significant.

Results

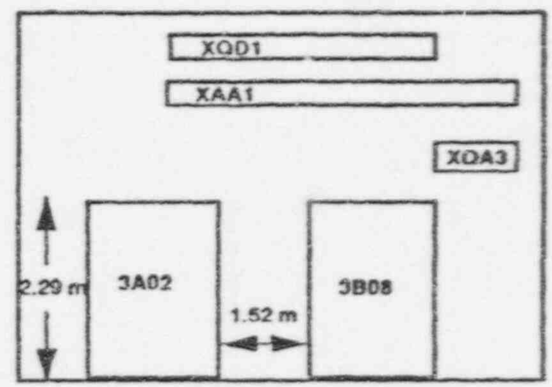
The Phase III CDF values for the above fire scenarios were added to obtain a revised overall CDF value of 2.3E-6/yr. As discussed in Section 4.5, the fire compartment will be evaluated for containment heat removal and isolation capability using the results of the IPE Level 2 Analysis.



LAYOUT DRAWING: 3-PE-63-3B
ELECTRICAL PENETRATION ROOM



ELEVATION : 3-PE-63-3B



SIDE : 3-PE-63-3B

FIGURE 4.4-8
WALKDOWN RESULTS FOR FIRE COMPARTMENT 3-PE-63-3B

4.4.3 RESULTS OF PHASE III EVALUATIONS

Phase III was performed for each of the fire compartments which did not screen out in Phase II. Each fire compartment was assessed individually in order to evaluate significant contributors to fire risk, and to use the most appropriate means available to determine a realistic estimate of core damage frequency. Rather than assuming that all equipment and cables in a compartment would be lost, Phase III fire compartments were generally evaluated by subdividing the compartment fire sources and targets into scenarios. For each scenario, COMPBRN computer analyses were used to determine the probability and time to target damage from various fixed and transient fire sources. The unavailability of fire suppression was calculated for those scenarios where suppression could be credited. The conditional probability of core damage was then factored in for each specific fire scenario. The CDF for the resulting scenarios were summed to determine the total CDF for the compartment. The results are presented in Table 4.4-17.

TABLE 4.4-17
FIVE PHASE III RESULTS

FIRE COMPARTMENT	DESCRIPTION	APPROACH TO CDF REDUCTION	PHASE III CDF	SCREEN
2-PE-45-3A	Electrical Penetration	COMPBRN fire modeling	1.5E-6	No
2-PE-63-3B	Electrical Penetration	COMPBRN fire modeling	2.4E-6	No
2-AC-9-16	Corridor	Fault tree modeling	7.0E-8	Yes
2-AC-9-17	Relay Room	COMPBRN fire modeling	1.2E-6	No
2-AC-30-20A	Control Room	COMPBRN fire modeling, administrative change to AOI SO23-13-2,"Shutdown from Outside the Control Room"	8.0E-7	Yes
2-AC-30-20C	Computer Room	Evaluation, justification for no further analysis	8.3E-8	Yes
2-AC-50-35	Switchgear Room	Fault tree modeling	3.1E-6	No
2-AC-50-40	Switchgear Room	Fault tree modeling	3.4E-6	No

TABLE 4.4-17
FIVE PHASE III RESULTS

FIRE COMPARTMENT	DESCRIPTION	APPROACH TO CDF REDUCTION	PHASE III CDF	SCREEN
2-AC-85-70	Switchgear Room	COMPBRN fire modeling	8.1E-7	Yes
2-AC-85-71	Switchgear Room	COMPBRN fire modeling	1.1E-6	No
2-AR-63-116	Corridor	Fault tree modeling	3.0E-7	Yes
2-TB-148	Turbine Building	Fault tree modeling	2.3E-6	No
2-DG-30-155	Diesel Generator Room	Fault tree modeling, Administrative change to AOI SO23-13-21,"Fire"	1.2E-6	No
2-DG-30-158	Diesel Generator Room	Fault tree modeling, Administrative change to AOI SO23-13-21,"Fire"	1.2E-6	No
2-YD-30-200	Unit 2 and 3 Yard Area	Fault tree modeling	6.4E-7	Yes
3-PE-45-3A	Electrical Penetration	COMPBRN fire modeling	9.8E-7	Yes
3-PE-63-3B	Electrical Penetration	COMPBRN fire modeling	2.3E-6	No
Total CDF of Fire Compartments Not Screened By Phase III (Note 1)			1.6E-5/yr	

Note 1: This total CDF is derived from a cutset culling limit consistent with that of the Internal IPE (1.E-9/yr). CDF derived from this more rigorous calculation is slightly lower than the relatively conservative FIVE Phase III calculation. As a result, the total core damage frequency is lower than the summation of CDFs associated with the unscreened fire compartments.

The following fire protection features were important factors in reducing the calculated fire risk:

- Adequate physical separation of safety related equipment, cables, and ignition sources, allowing the evaluation of separate fire scenarios, such as the yard area and 63' radwaste corridor.
- Automatic suppression systems, such as in the auxiliary building corridor (9' elevation) and turbine building.
- Administrative changes to safe shutdown procedures to credit offsite power (if available), combined with manual fire suppression (control room), and power recovery to the 4 kV switchgear (diesel generator rooms).
- Manual suppression of electrical fires in the relay room.

The primary factors which contribute to the higher fire risk in these Phase III fire compartments are:

- Limited separation of Unit 2 and Unit 3 offsite power equipment and cables, such as in the relay room and penetration areas.
- Fusing configuration of Unit 2 and Unit 3 offsite power circuits, causing potential loss of offsite power to both units due to a fire in the 6.9 kV non-essential switchgear.
- Conservative estimate of diesel generator unavailability to run, leading to relatively high conditional probability of core damage associated with loss of one train of essential power.
- Relatively high fire frequencies for the turbine building fire scenarios, combined with the conditional core damage probability associated with loss of main feedwater.
- The circuit design of the diesel generator feeder breaker to the 4 kV essential switchgear, causing potential loss of power to the switchgear due to a fire in the diesel generator rooms. Operator recovery actions are available for mitigation of this scenario.

Of the 17 fire compartments which were not screened in Phase II, an additional 7 compartments were screened in Phase III. The remaining 10 compartments have a total CDF of 1.6E-5/yr. The next section provides an analysis of the containment performance and level 2 impacts for these fire scenarios.

4.5 ANALYSIS OF CONTAINMENT PERFORMANCE

Generic Letter 88-20, Supplement 4, Section 4.2 directs the following:

> "Containment performance should be assessed to determine if vulnerabilities stemming from sequences that involve containment failure modes distinctly different from those obtained in the internal events analysis are predicted."

In Appendix 2 of Generic Letter 88-20, Supplement 4, the following aspects of this assessment are listed:

- "Identify mechanisms that could lead to containment bypass"
- "Identify mechanisms that could cause failure of the containment to isolate"
- "Determine the availability and performance of the containment systems under the external hazard to see if they are different from those evaluated under the internal event hazard evaluation"

The plant walkdowns for the fire IPEEE (discussed in Section 4.1.3) did not result in the identification of any additional or unique fire-related containment failure modes. The fire events are therefore modeled the same as an internal event with regard to containment response, but include the equipment losses due to the fire.

The potential for containment bypass in case of fire is addressed in Section 4.5.1. The potential impact of fires on containment isolation is addressed in Section 4.5.2. The evaluation results of the potential fire impacts on containment systems are discussed in Section 4.5.3 and the potential for release of radioactivity from the plant in the event of a fire is discussed in Section 4.5.4.

4.5.1 CONTAINMENT BYPASS POTENTIAL

The potential for a fire-induced interfacing systems LOCA (ISLOCA) was evaluated by reviewing the Appendix R examination of high-low pressure interfaces, (including the closed loop shutdown cooling lines, the CVCS letdown lines, and the pressurizer head vents). The cables for the isolation valves for piping with these high-low pressure interfaces were identified for each fire compartment, as listed in the Phase I analysis documentation. The valves in each line configuration were checked to verify that a fire would not cause failure of all isolation valves, including the potential for hot shorts. Based on the cable separation, and valve power and control configuration, it was concluded that these ISLOCA scenarios could be screened from further analysis.

The other bypass potential evaluated for the fire IPEEE is a fire resulting in a steam generator tube rupture (SGTR). A fire cannot directly cause a steam generator tube to rupture. All core damage sequences, however initiated, that progress with a high primary system pressure have a small possibility of the consequent occurrence of an "induced" SGTR. This would be caused by failure of a tube under the combined effects of very high temperatures from the melting core off-gases and high differential pressures. This is not peculiar to fire-initiated sequences however and the dependent probabilities are the same as for internal event-initiated sequences. These induced SGTRs are considered in the containment response discussed in Section 4.5.4.

4.5.2 Containment Isolation Analysis

The internal events IPE containment isolation analysis was used as the basis for the fire IPEEE containment isolation analysis. Since containment isolation is not one of the safety functions included in the Appendix R analysis, the associated valves and cables were not explicitly included in the Appendix R analysis. Therefore, since only a few of the IPE containment isolation valves were included in the Appendix R Safe Shutdown Equipment list, a special review was performed to ensure that a fire would not cause a unique failure of the containment isolation function. Specific cable tracing was not performed.

The containment isolation fault tree from the IPE was directly used in the extended accident sequence event trees for the fire level 2 analysis. Power dependencies, including direct fire losses as well as random failures, for the containment isolation valves were directly included in the fault tree quantification. Each containment penetration and the associated valves included in the IPE were evaluated to verify that loss of power and control cables would not fail the isolation function. Most of the penetrations had at least one isolation valve that would fail to the closed position if the cables were lost in the fire. Some penetrations included check valves, whose function would not be affected by a fire. The remainder of the isolation valves were associated with piping systems which are closed systems inside containment, and would therefore not need to be isolated for a fire-initiated event. The containment and containment purge isolation actuation signals (CIAS and CPIS) are fail-safe such that fire damage to cables would result in actuation of containment isolation. Thus, the impacts of the fire scenarios on the containment isolation function have been evaluated and included in the containment performance analysis.

4.5.3 CONTAINMENT SYSTEMS PERFORMANCE

The fire compartment analyses reported in Section 4.4 can be modeled by one of two classes of accidents that are counterparts to ones occurring in the internal events analysis with the additional constraint that systems lost directly by fire cannot be recovered. These classes are either a loss of offsite power or a loss of the power conversion system. Because of subsequent random failures, the loss of offsite power class may lead to a station blackout event.

Therefore, the effects of fires on containment systems with regard to core damage accidents can be modeled by estimating the frequency of occurrence of fire-induced initiating events of these classes, incorporating the specific fire losses for these scenarios, and then evaluating the results using internal event Level 1 and Level 2 models and data. The first two of these steps has been reported in Section 4.4. The third step, determining the containment response for these conditions, is presented in Section 4.5.4.

For reference the bounding initiating events are:

- Loss of Power Conversion System (PCS) events caused by fires identified as:
  - 2-AC-50-35 SWGR ROOM 2B
  - 2-AC-50-40 SWGR ROOM 2A
  - 2-TB-148 TURBINE BUILDING
  - 2-DG-158 DG ROOM TRN A
  - 2-DG-155 DG ROOM TRN B

- Loss of Offsite Power (LOOP) events caused by fires identified as :
  - 2-PE-45-3A PENE ROOM
  - 2-AC- 9-17 RELAY ROOM
  - 2-PE-63-3B PENE ROOM

3-PE-63-3B PENE ROOM
2-AC-85-71 SWITCHGEAR ROOM

- Station Blackout type events subsequent to the Loss of offsite power events listed above.

4.5.4 CONTAINMENT RESPONSE TO FIRE-CAUSED EVENTS

The SONGS 2/3 containment response assessment for fire events found that:

1. There are no new containment failure modes.
2. Fires can be considered as resulting in one of two types of initiating events that have already been modeled in the internal events IPE.

These results allow the containment response to be evaluated in the same way as the corresponding internal initiating events.

Similar to the IPE, these dominant fire damage scenarios were processed through extended event trees, plant damage states and containment event trees. Since the submittal of the IPE, the Level 2 PRA methodology has been upgraded to more directly include estimates of the effects of phenomenological uncertainties on quantification. This expanded Level 2 methodology retains the fundamental SONGS-specific phenomenological understandings presented in the IPE submittal and in responses to NRC Request for Additional Information (RAI). It is based on the modified REBECA software package for the level 2 numerical quantifications. In addition, NUCAP+ [4-12] software was also employed to solve intermediate steps in containment phenomenology.

The following containment response outcomes, in percentages of the fire-induced core damage frequency reported in Section 4.4.3, have been determined from the quantitative Level 2 assessment. For comparison, the IPE Level 2 results are also included in Table 4.5-1.

The containment "not failed" category (82% of the total fire IPEEE results as stated in Table 4.5-1) contains the fire-induced core damage sequences that do not involve over-pressurization failure of containment. However, sequences with the potential of late basemat melt-through from core concrete interactions are included in this category if long-term recovery actions are not credited. The percent of fire IPEEE sequences that have the potential of late basemat melt-through is consistent with the IPE results. The

TABLE 4.5-1
DISTRIBUTION OF FIRE IPEEE AND IPE
CONTAINMENT PRESSURE BOUNDARY STATUS

CONTAINMENT PRESSURE BOUNDARY STATUS	FIRE IPEEE RESULTS	INTERNAL EVENT IPE RESULTS
Containment Not Failed	82%	83%
Leak Type Containment Failure	11%	8%
Rupture Type Containment Failure	5%	4%
Steam Generator Tube Ruptures (Initiated or Induced)	2%	3%
Containment Bypassed (ISLOCA)	None	2%

timing of the occurrence of the containment failure categories for the leak and rupture failures modes (excluding the induced SGTR) is:

Time of Containment Failure:	
At or about the time of Vessel Failure	1%
Late (more than 12 hours later)	15%

The release of radioactivity, using the release categories defined in Section 4.8 of the internal events IPE report [4-4], is distributed among the categories identified in Table 4.5-2.

TABLE 4.5-2
RELEASE CATEGORY AND PROBABILITY OF FIRE IPEEE

RELEASE CATEGORY	RELEASE CATEGORY DEFINITION	RELEASE FREQUENCY (PER YEAR)	PERCENT OF TOTAL FIRE CORE DAMAGE FREQUENCY
S	Success, no containment failure within 48 hours, < 0.1% volatiles released	1.4E-5	82%
L	Late containment failure, up to 1% volatiles released	2.4E-6	15%
B	Containment bypassed, < 0.1% volatiles released	2.3E-7	1%
W	Late containment failure, more than 10% volatiles released	2.0E-7	1%
G	Early/isolation failure, containment failure prior to/at vessel failure, up to 10% volatiles released	1.4E-7	1%
D	Containment bypassed, up to 10% volatiles released	0	0
T	Containment bypassed, > 10% volatiles released	0	0

The single most important factor for the small release Category S is arrest of the core damage process before vessel failure due to the timely availability of injection and containment heat removal. The majority of these sequences are loss of power conversion system (PCS) with loss of all feedwater with ECCS available. The most significant cut set (at 4%) has core damage caused by operator failure to provide long term CST makeup after power conversion system (PCS) loss due to a fire at location 2-TB-148 in the Turbine Building. The next two most significant cut sets (at 2% each) involve failure of the motor driven auxiliary feedwater pump to run for 24 hours, with the turbine driven auxiliary feedwater pump out for maintenance, after a LOOP caused by fires in either Switchgear Room 2A or 2B (2-AC-50-40 or 2-AC-50-35). In these three scenarios the core starts to melt with the RCS at high pressure, with the likely consequence of inducing the hot leg piping to fail. This depressurizes the RCS and allows injection to begin, which arrests the core melt in vessel and limits the source term to a small "S" release.

Station blackout (SBO) class sequences, resulting from random failure of all Unit 2 and 3 diesel generators to supply emergency power after a fire-induced LOOP, contribute two-thirds of the Category L (typically late failures with dry cavities) releases. Loss of PCS sequences with no auxiliary feedwater, no injection, and no containment heat removal, contribute another one quarter this category. The two most significant cut sets (each at 4% of the total) are SBO events caused by random failures to run for one day of all four diesel generators following a LOOP from a fire in the 2-AC-85-71 switchgear room and 2-AC-9-17 relay room. The next two most significant cut sets, (at 3% each of the total), are SBO events caused by common cause failure of all diesel generators after a LOOP event due to a fire in either of the 2-PE-63-3B or 3-PE-63-3B penetration rooms. The containment failure mode for these scenarios is a late overpressure failure that is more likely to be a leak than a rupture after induced hot leg and vessel failures with the debris dropping into an unflooded cavity.

Approximate two-thirds of both the Category B release frequency and the Category W release frequency are due to loss of PCS sequences with failure of all feedwater, where injection and containment heat removal are available in the core damage phase. Category B is primarily the induced SGTR releases where the secondary relief valves function normally, and Category W is late containment failures with the cavity flooded. The most significant cut set (at 4%) has core damage caused by operator failure to provide CST makeup after power conversion system (PCS) loss due to a fire at location 2-TB-148 in the turbine building. The next two most significant cut sets (at 2% each) involve failure of the motor driven auxiliary feed pump to run for 24 hours, with the turbine driven pump out for maintenance, after a LOOP caused by fires in either switchgear room 2A or 2B (2-AC-50-40 or 2-AC-50-35). In these three scenarios the core starts to melt with the RCS at high pressure, with the likely consequence of inducing the hot leg piping to fail. This depressurizes the RCS and allows injection to begin, but does not arrest the core melt in vessel. The core debris subsequently is retained in a flooded cavity. The containment fails late with the debris overflooded and which results in a "W" category source term release. The dominant cut sets for the Category "B" releases are the same as that for the Category "W" except that a steam generator tube rupture is induced by very high RCS pressures and temperatures. The cut set contribution fractions are slightly different (6%,2%,2%) because of the different overall sequences contributing to the classes.

Category G (early) releases are due to phenomenological assessments of events such as hydrogen burns or direct containment heating (DCH) occurring at about the time of vessel failure for all the core damage sequences. The four most significant cut sets (at about 2% each) are the same as those listed for "L" releases above, except that an early containment failure is predicted rather than a late failure. The frequency of this release category is higher than the IPE results due to the consideration of low probability events (e.g., hydrogen burn or DCH that might contribute to early containment failure) in the expanded Level 2 method.

One major insight that can be drawn regarding the Level 2 effects consequent to fires is that the frequency of fire-induced bypass sequences is low. The frequency is low both absolutely and as a fraction of the core damage frequency when compared to the internal events IPE results (Table 4.5-1). The reasons are that there are no interfacing systems LOCAs or SGTRs directly caused by fires, and that the frequency of isolation failure is calculated to be negligible.

Another important insight is that the proportion of no containment failure outcomes is approximately the same for the internal events and the fire-induced events. That is, more than 80% of the core damage sequences do not result in containment atmospheric pressure boundary failure. Most of the containment failures that do occur are late failures. The majority of the containment failures that do occur are leaks rather than ruptures. These results are also consistent with the internal event results.

4.6 TREATMENT OF THE FIRE RISK SCOPING STUDY ISSUES

The purpose of this evaluation is to address potential contributors to fire risk identified by Sandia National Laboratory in their Fire Risk Scoping Study [4-5]. Based on Sandia's review of four fire PRAs, there were six potential contributors to fire risk which had not been adequately addressed in these older risk assessments. This section evaluates these issues for SONGS 2/3 using the guidelines presented in the FIVE Methodology, Section 7 and Attachment 10.5 [4-1].

The NRC staff has requested that the following six issues be addressed in future fire evaluation methodologies:

1. Seismic / Fire Interactions
2. Fire Barrier Qualifications
3. Manual Fire Fighting Effectiveness
4. Total Environment Equipment Survival
5. Control Systems Interactions
6. Improved Analytical Codes

Each issue was evaluated for SONGS 2/3 and is discussed below.

4.6.1 SEISMIC / FIRE INTERACTIONS

This issue involves three concerns: (1) seismically induced fires, (2) seismic actuation of fire suppression systems, and (3) seismic degradation of fire suppression systems. These concerns are addressed in the IPEEE evaluation for seismic events, Section 3.3.4. The analysis is based primarily on plant walkdowns performed to identify potential problems associated with seismic / fire interaction. No vulnerabilities were identified.

4.6.2 FIRE BARRIER QUALIFICATIONS

Fire barrier qualifications for SONGS 2/3 were evaluated using the format provided in Section 10.5 of the FIVE Methodology. The SONGS 2/3 fire protection program was assessed against the attributes of an adequate fire protection program discussed in FIVE. Based on this assessment, it was determined that fire barrier qualifications were adequate to ensure mitigation of potential fire risk.

As discussed in the following sections, the operability of fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. SONGS 2/3 fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers and fire doors are periodically inspected to verify their operability.

Fire Barriers

Technical Specification Section 3.7.9 and Surveillance Requirement 4.7.9 contain surveillance requirements for fire rated assemblies, which include: fire barriers, fire doors, fire windows, fire dampers, seismic gap seals and cable, ventilation duct and piping penetration seals.

Fire Doors

Technical Specification fire door inspection requirements are outlined in Surveillance Requirement 4.7.9 as follows:

1. Verify at least once per 24 hours the position of closed fire doors, and that doors with automatic hold-open and release mechanisms are free of obstructions.

2. Verify the position of locked closed fire doors at least once per 7 days.

3. Perform a channel functional test of the fire door supervision system at least once every 31 days.

4. Inspect the automatic hold-open, release and closing mechanisms and latches at least once every 6 months.

5. Perform a functional test of the automatic hold-open, release and closing mechanisms and latches at least once every 18 months.

6. Perform a visual inspection of the exposed surfaces of each fire rated assembly at least once every 18 months.

Inspection and maintenance methods and schedules are provided in SONGS procedures SO23-XIII-50 and SO23-XIII-58.

Penetration Seal Assemblies

Penetration seal inspection requirements are outlined in Surveillance Requirement 4.7.9 as follows:

> "At least once every 18 months, perform a visual inspection of at least 10% of each type (mechanical & electrical) of sealed penetration. If any apparent changes in appearance or abnormal degradation is found, and inspection of an

additional 10% shall be made. This process will continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected so that each penetration seal will be inspected at least once every 15 years."

Inspection and maintenance methods and schedules are provided in SONGS procedures SO23-XIII-02 and SO23-XIII-57.

In response to the expressed concerns in NRC Information Notice 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals," a penetration seal evaluation program was developed at SCE for SONGS Units 2 & 3. This program, entitled "Fire Area Boundary Penetration Seal Evaluation Program," evaluated the installation and qualifications of penetration seals located in barriers that protect safe shutdown equipment or are otherwise required by BTP 9.5-1. This included 9,749 of approximately 16,000 seals installed at SONGS 2 & 3. The purposes of the Fire Area Boundary Penetration Seal Program were: (1) to ensure that the penetration seals installed in fire barriers at SONGS 2 & 3 were adequate to perform their intended function, and (2) to implement corrective action, where necessary, to ensure the integrity of required fire barriers. The results of the program are as follows:

a. The majority of seal detail designs (26 out of 31) were backed by valid fire endurance tests.

b. In cases where the seal detail deviated significantly from the tested seal configuration, seal-specific acceptance criteria were developed to define the situations under which the seal detail is supported by tests.

c. The construction process, including procedures, training, design and material control, was adequate to assure that the installed seal met the original design intent.

d. In the vast majority of cases, the installed seals followed the appropriate seal detail and were built within the limits of the acceptance criteria. All seals requiring an IN 86-10 evaluation were determined to be qualified for use in their associated barriers.

e. Walkdowns revealed a very low proportion of "as-built" conditions which differed substantially from designs.

   - 17 damaged seals were identified to Emergency Preparedness personnel for corrective action.

- All identified deviations were reviewed and found to be acceptable.

f. Documentation was annotated and corrected where necessary to reflect the "as-built" configuration.

g. All 9,749 seals reviewed in this program were determined to be rated consistent with their associated barrier or qualified for use in that barrier.

The Fire Area Boundary Penetration Seal Evaluation Program concluded that the penetration seals installed at SONGS Units 2 & 3 are adequate to perform their intended function in the fire barriers in which they are installed.

Fire Dampers

Fire damper inspection requirements are outlined in Surveillance Requirement 4.7.9, and include performance of a visual inspection of all fire dampers and associated hardware at least once every 18 months. Inspection and maintenance methods and schedules are provided in SONGS procedure SO23-XIII-57.

NRC Information Notice 83-69, "Improperly Installed Fire Dampers at Nuclear Power Plants," addressed concerns regarding proper installation and ratings of fire dampers. SCE has administrative controls and procedures to ensure that damper installations and damper ratings conform with design requirements. Walkdowns performed for the 1984 Fire Hazards Analysis Update verified that the fire dampers were of the required fire rating.

NRC Information Notice 89-52, "Potential Fire Damper Operational Problems", expressed concerns about potential problems affecting the closing reliability of curtain-type fire dampers under ventilation system operational air flow conditions. An evaluation of this issue was documented in the SONGS 2/3 Fire Damper Study - Response to NRC Information Notice 89-052. This document determined that current test procedures (SO23-XIII-57) did not ensure establishment of "worst-case" flow conditions prior to or during testing, and concluded that proper performance of these dampers in a fire was only assured if the ventilation systems to an area were administratively shut down upon confirmation of a fire. This method, combined with testing under air flow conditions if necessary, will be utilized to ensure fire damper closure. The following procedural changes were implemented in regards to these requirements:

a. Revise SONGS Units 2 & 3 Time and Manpower analyses to include manual actions required for each fire area to de-energize fans.

b. Revise SONGS Units 2 & 3 operator instructions to include manual actions required for each fire area to de-energize fans.

c. Revise SONGS Units 2 & 3 surveillance procedure SO23-XIII-57 to require testing under airflow conditions, as necessary, in cases where manual actions may not be feasible.

Completion of the recommended tests and any required modifications, in addition to these changes, will ensure that all technical specification required dampers close when exposed to fire.

4.6.3 MANUAL FIRE FIGHTING EFFECTIVENESS

The ability of the fire protection program to provide effective manual fire fighting was assessed using the format contained in the FIVE Methodology Section 10.5. It was determined that the SONGS 2/3 fire protection program satisfies the necessary attributes which ensure adequate manual fire suppression response thus allowing reliance on manual suppression in the fire risk evaluation. These attributes are discussed below.

Reporting Fires

1. Appropriate plant personnel are knowledgeable in the use of portable fire extinguishers. The San Onofre Fire Departments consists of professionally trained, full time personnel whose primary responsibility is fighting fires at San Onofre. These individuals are trained per Procedure S0123-XIII-20, "Fire Department / Emergency Services Officers Training Program." One of the primary objectives of this procedure is to train fire department personnel to manually suppress fires utilizing the various fixed extinguishing systems, portable extinguishers, hose lines, and fire apparatus that exist at the station.

2. Per the UFHA, portable fire extinguishers for manual extinguishing of fires are provided throughout the plant in accordance with the recommendations of NFPA

10 - 1975. Extinguisher approximate locations are shown on the feature drawings and are listed in the extinguisher surveillance procedure, SO123-XIII-52.

3. Plant Procedure SO123-XV-4.10 "Fire Fighting" exists for reporting fires in the plant. Section 6.1 of this procedure instructs all personnel on emergency notification in the event of a fire. This includes phone numbers for contacting Emergency Services and the fire department.

4. A plant communication system exists that includes contact to the control room. A telephone system, an intercom, UHF radio, and a public address system are provided to accomplish onsite communication between the control room and various plant locations under normal and emergency conditions. Communication will be available from the fire staging areas via a sound power phone system evaluation in Document 90035BF, "Sound Powered Phones."

Fire Brigade

1. The San Onofre Fire Department is certified by the State of California Fire Marshal's office to provide fire suppression activities. Per the UFHA, a minimum of five certified fire fighters are on duty per shift.

2. Per the UFHA, to ensure adequate breadth of knowledge in plant systems during all fire emergencies, the affected control room will respond with an Assistant Control Operator or equivalent to coordinate with the Fire Department Chief and act as a technical advisor.

3. All fire department personnel shall successfully pass the standard SCE pre-employment medical exam. On an annual basis, personnel should successfully pass the Haz-Mat physical exam conducted by the SCE Medical Department. These requirements are proceduralized in SO123-XIII-20.

4. As documented in the UFHA, the equipment necessary for effective fire department performance are provided. This includes personal protective equipment such as SCBA, turnout coats, boots, gloves, and hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers.

Fire Brigade Training

Per the UFHA, all fire department members receive training which includes fire fighting with portable extinguishers, the use of hose lines, ventilation of buildings, salvage operations, rescue operations, and the special aspects of fighting fires in controlled areas. It also covers the use of new equipment, procedures, methods and hazards. The training program is consistent with the attributes of an adequate fire protection program and is documented in Procedure SO123-XIII-20.

Practice

Per Procedure SO123-XIII-20, practice sessions shall be held a minimum of once a year to provide "hands on" experience in fire extinguishment. This includes the use of emergency breathing apparatus under strenuous conditions associated with various types of fires that could occur at the plant.

Drills

Fire department drills, performed per Procedure SO123-XIII-21, are consistent with the attributes of an adequate fire protection program described in FIVE. Treatment of fire pre-plans per Procedure SO123-XIII-5, and maintenance of all fire department equipment are also adequate.

Records

Training records for all fire department personnel are completed and maintained in accordance with the SONGS Training and Records Maintenance Program as discussed in Procedure SO123-XIII-20.

4.6.4 TOTAL ENVIRONMENT EQUIPMENT SURVIVAL

Total equipment survival against the adverse effects of combustion products and inadvertent fire suppression for SONGS 2/3 was evaluated using the format provided in Section 10.5 of the FIVE Methodology. Operator effectiveness in performing manual safe shutdown actions and potential misdirected suppression effects in smoke filled environments were also considered. The SONGS 2/3 fire protection program was assessed against the attributes of an adequate fire protection program discussed in FIVE. Based on this assessment, and taking into account guidance provided in the FIVE Methodology, it was determined that total equipment survival was adequate to ensure mitigation of potential fire risk.

Potential Adverse Effects on Plant Equipment by Combustion Products

As discussed in the FIVE Methodology, there have not yet been enough studies performed with respect to non-thermal fire effects on industrial plant equipment to adequately quantify the potential problems and identify solutions each utility should consider for those problems. SONGS Units 2 & 3 has established a post-fire smoke removal process, in part to alleviate the potential adverse effects of smoke, which was reviewed and approved by the NRC in May of 1985. The FIVE Methodology does not currently allow for an evaluation of non-thermal environmental effects of smoke on equipment. However, per the FIVE Methodology, the detrimental short term effects of smoke on equipment are not believed to be significant.

Spurious or Inadvertent Fire Suppression Activation

NRC Information Notice 83-41, "Actuation of Fire Suppression Systems Causing Inoperability of Safety-Related Equipment" [4-13], expressed several concerns regarding the inadvertent actuation of suppression systems causing damage to equipment credited for safe shutdown. Several actions were initiated to respond to these concerns. The conclusions of the 83-41 analysis are as follows:

1. Contamination of diesel fuel oil by fire suppression system water will not occur at SONGS 2 & 3, as the fire suppression system is not connected to the diesel fuel oil tank.

2. Most plant areas are provided with floor drains sized to remove expected fire fighting water. An analysis of water accumulation in plant areas without drains sized for fire protection systems has demonstrated that safe shutdown capability will not be adversely impacted by flooding.

3. Actuation of the water suppression systems at SONGS 2 & 3 due to inadvertent actuation of smoke detectors will not occur, as the alarm and actuation systems are separate. Smoke detectors are used for alarm and heat and flame detectors are used for actuation of the suppression systems.

4. Problems related to improper or inadequate design as expressed in IE Notice 83-41 are not applicable to SONGS 2 & 3. The deluge valves are electrically operated by a solenoid valve that needs to be energized. Therefore, a loss of power will not actuate the deluge valves. In the event of a loss of power, backup power is available for actuating the deluge valves, if necessary.

5. A study of the impact of inadvertent actuation of the water suppression system on safety related equipment was performed. All areas of the plant containing both automated water suppression systems and safety related equipment were considered in this analysis. The flooding analysis was performed assuming inadvertent operation of the water suppression system for 30 minutes, and flood heights were calculated from known water discharge rates and floor areas. The results of this analysis found that, in general, inadvertent operation of the water suppression system would not impact the safe shutdown capability of the plant. Recommendations of this analysis included: (1) Installation of 9" curbs and weatherstripping between the cable riser gallery and control room doors, and (2) installation of weatherstripping on the doors between the switchgear rooms and the cable riser galleries. These changes will minimize the level of flooding in the control room and switchgear rooms due to inadvertent actuations in the cable riser galleries, and were implemented in design change package 952.1M.

Operator Action Effectiveness

1. Abnormal Operating Instruction SO23-13-21, "Fire," provides instruction on normal safe shutdown procedures in the event of a fire. Abnormal Operating Instruction SO23-13-2, "Shutdown from Outside the Control Room," provides instruction on alternate safe shutdown procedures in the event of a fire in the control room.

2. Classroom and simulator training on abnormal and emergency operating procedures for operators is performed per procedure SO123-XXI-1.11.7, "Licensed Operator Retraining Program". This procedure also requires that all abnormal and emergency operating procedures shall be covered on a biennial basis. Operators are advised of changes to operating procedures via the Operations Summary Report. This report is a weekly overview of changes to operations procedures and it is required by procedure SO123-0-34, "Distribution and Acknowledgment of Information".

3. If necessary in the performance of these procedures, self-contained breathing apparatus (SCBA) equipment, using full-faced positive pressure masks with a minimum of one half-hour service, is provided for fire brigade, damage control and control room personnel. In addition, two extra air bottles are supplied for each SCBA unit. Procedure SO123-XIII-9 provides instructions for the inspection, wearing and use of SCBA equipment.

4.6.5 CONTROL SYSTEMS INTERACTIONS

Control systems interactions for SONGS 2/3 were evaluated using the format provided in Section 10.5 of the FIVE Methodology. The SONGS 2/3 fire protection program was assessed against the attributes of an adequate fire protection program discussed in FIVE. Based on this assessment, it was determined that control systems interactions were designed adequately to ensure mitigation of potential fire risk.

Safe shutdown circuits required for alternate shutdown of SONGS Units 2 and 3 can be isolated from the control room in the event of a fire in the control room area. This capability is examined in Appendix R Calculation 90035AK, Alternative Shutdown Capability Evaluation. This document identifies the systems and components required for alternate shutdown and evaluates whether they can be isolated from the control room. For components with isolation switches which did not provide adequate isolation from the control room, it was recommended that the isolation switch be rewired (modification calculation 90035AJ). Additional isolation switches were recommended where required for components without isolation switches (modification calculation 90035AJ). These modifications were performed for Units 2 and 3 in Design Change Packages 2-6554 and 3-6554.

4.6.6 IMPROVED ANALYTICAL CODES

This issue involves questions regarding the adequacy of available fire models for use in IPEEE analysis for fire external events. Per the FIVE Methodology, after a number of discussions between nuclear industry representatives and the NRC staff regarding this issue, the NRC agreed that the COMPBRN IIIe fire modeling program is adequate for analytical fire modeling and application in IPEEE analysis. The fire modeling techniques incorporated in Phase II of the FIVE Methodology are derived from the same basic correlations used in COMPBRN IIIe. In addition, the SONGS 2/3 Phase III evaluation uses COMPBRN IIIe to further analyze fire risk. No additional evaluation is required for this issue.

4.6.7 CONCLUSION

Based on this evaluation, the issues presented in the Sandia Fire Risk Scoping Study were found to have no unanalyzed impact on fire risk at SONGS Units 2 and 3.

4.7 USI A-45 AND OTHER SAFETY ISSUES

This section discusses the following NRC safety issues with respect to fire risk:

1.	USI A-45	Decay heat removal
2.	SECY-93-143	Reassessment of the NRC Fire Protection Program

4.7.1 USI A-45: DECAY HEAT REMOVAL

The USI A-45 issues is concerned with reliability and potential vulnerabilities in the decay heat removal systems, both for internal and external events. For SONGS 2/3, the safety-related decay heat removal systems for the A-45 issue include the auxiliary feedwater system, the high and low pressure safety injection systems, and the containment spray system. Support systems may include electric power, cooling water (chilled water, CCW and SWC), air/nitrogen, and room cooling and ventilation.

For the case of a transient or small LOCA, the AFW system removes decay heat through the steam generators, either to the main condenser, or to the atmosphere through the atmospheric dump valves or secondary side safety valves. Long-term decay heat removal is provided through the closed loop residual heat removal system, utilizing the LPSI pumps and shutdown cooling heat exchangers. In case of a LOCA, the HPSI, LPSI, and CS systems can provide primary inventory makeup and decay heat removal during recirculation. Containment heat removal is also available from the containment fan coolers for the LOCA events.

Each of these systems were included in the analysis of potential fires for the IPEEE. Based on the relatively low CDF from fires at SONGS 2/3, and the conservatism included in the fire modeling process, there are no identified vulnerabilities in the decay heat removal systems. Generally, the sequences leading to potential core damage involve the fire initiating event, and multiple additional failures, including failure of potential operator recovery actions. While the AFW system is very important for decay heat removal, the system components and cables are well-separated and protected from potential fires. Also, the three AFW pumps have diverse power sources which are well-protected from potential fires. Other support systems have similar protection. While a fire-induced LOCA would be very rare, it was analyzed in the fire IPEEE analysis. The HPSI, LPSI, CS system, and support systems are redundant-designed systems with fire protection features. If a fire were to cause control room evacuation, alternate shutdown can be accomplished through the alternate shutdown control panels. In addition, the fire procedures provide detailed and well-practiced actions for provision of decay heat removal in the event of fires.

In summary, a plant-specific systematic evaluation has been performed for SONGS 2/3 to identify any potential vulnerabilities in the decay heat removal systems. No vulnerabilities were identified for fire initiating events.

4.7.2 EVALUATION OF NRC POLICY ISSUE SECY-93-143

This section addresses NRC Policy Issue SECY-93-143 dated May 21, 1993 (Subject: NRC Staff Actions to Address the Recommendation in the Report on the Reassessment of the NRC Fire Protection Program). As discussed in the NRC report, certain issues identified during the course of the NRC review effort are of a plant-specific nature and if possible, should be addressed through the IPEEE program. These issues, listed as IPEEE follow-up issues in Table 10.2-3 of the NRC report, identify some items discussed in the Sandia Fire Risk Scoping Study Evaluation. These are: (1) protection from control systems interactions (including identification and resolution of "risk-significant" fire scenarios), (2) equipment protection from fire suppression system actuation, (3) potential vulnerabilities due to broken or leaking flammable gas lines, and (4) potential vulnerabilities due to seismic/fire interactions. The first issue was discussed previously in Section 4.6.5. The other three issues were discussed in the seismic IPEEE Section 3.3.4.

Other issues listed in the NRC report are discussed below.

Adequacy of the FIVE Methodology

In the NRC assessment of the FIVE Methodology, concerns were raised that the methodology may not identify all of the significant accident sequences related to fire events. This potential oversight was attributed primarily to the Phase I screening criteria for fire areas. The initial FIVE Methodology allowed fire areas to be screened if they did not contain Appendix R safe shutdown equipment. This screening criteria could overlook areas which contain primary contributors to core damage frequency (e.g., loss of off-site power). This and other issues associated with Phase I of the FIVE Methodology were addressed in the Phase I Revision 1 changes, and documented in Section 4.2.1.9.

Potential Vulnerabilities due to Fire Shutdown System Criteria

Exposure to non-conservatism could result if 1) shutdown systems are not required to satisfy single failure, or other criteria including non-fire related failures concurrent with fire events or seismic criteria, and 2) plant accidents or natural phenomena are not postulated concurrently with fire events. These NRC issues are addressed below as they apply to SONGS 2 and 3.

1. Single failure and other non-fire related failure criteria: The probability of failure of shutdown systems due to single failure and other non-fire related failures is inherent in the SONGS 2/3 IPEEE model. This aspect of the model is applied to Phase II fire compartments in order to identify potential vulnerabilities due to fire. For Phase I fire compartments which are not analyzed in Phase II, a revised method is presented in the Phase I calculation to address potential vulnerabilities. In all cases where screened fire compartments contain shutdown equipment, potential vulnerabilities are addressed by considering the reliability of the shutdown equipment (using the IPE model) or by bounding the fire compartment by a Phase II fire compartment with equal or greater risk significance. This is discussed further in the SONGS 2/3 Phase I Revision 1 changes, Section 4.2.1.9. Notably, single failure criteria applied to safety systems is less limiting than Appendix R failure criteria applied in the event of fire since Appendix R failure criteria postulates multiple simultaneous failure of a larger set of safety system components.

2. Seismic criteria: A majority of SONGS 2/3 shutdown systems are required to satisfy seismic criteria. Potential vulnerabilities due to seismic/fire interactions are included in the seismic evaluation, Section 3.3.4.

3. Plant accidents or natural phenomena: The initiating event frequency of these types of accidents in combination with a fire scenario, is small enough to offset any increase in risk due to additional accident-related safe shutdown equipment failures. Fire-induced risk is normally assessed based on a 24 hour mission time such that a concurrent event would be unlikely. Furthermore, the risk from natural phenomena and nearby facilities assessed in Section 5 concluded that the risk posed by these external events is not significant. The contribution to fire risk due to plant accidents and natural phenomena, therefore, is judged to be acceptably small.

Based on this evaluation, the issues presented in SECY-93-143 were found to have no unanalyzed impact or potential vulnerability for fire risk at SONGS Units 2 and 3.

4.8 SUMMARY OF INTERNAL FIRES ANALYSIS

The internal fire analysis fulfills the objectives of the IPEEE, and provides a systematic examination to identify any plant-specific vulnerabilities to severe accidents initiated by internal fire events.

In conformance with NRC GL 88-20, Supplement 4, and NUREG-1407, the internal fire analysis used a combination of the two NRC-approved approaches, FIVE and fire PRA. The EPRI FIVE methods [4-1] were used for progressive screening of most fire compartments, and more detailed COMPBRN fire modeling [4-2] and PRA methods [4-3] were used for the analysis of non-screened compartments.

Phase I Qualitative Screening Analysis

In Phase I, fire areas were screened out from further analysis if they did not contain Appendix R safe-shutdown components (or cables) and if a fire in that area would not cause a demand for safe-shutdown functions. Note that FIVE Revision 0 allowed screening if either of these criteria were met. The original Phase I analysis was performed before Revision 1 was issued. Rather than re-performing the Phase I analysis using the Revision 1 criteria, which would have reduced the number of areas screened out, a bounding quantitative analysis was performed to ensure that the original qualitative screening was acceptable. Fire compartments, which are subsets of fire areas, were similarly screened out if they met the above criteria, and had no potential for fire spread to other fire compartments.

Phase II Simplified Fire Compartment Analysis

The 85 fire compartments not screened out by the qualitative screening were evaluated using FIVE Phase II quantitative fire modeling. Fire risk was divided into three independent, multiplicative factors: the fire initiating event frequency, the unavailability of redundant or alternative equipment not affected by the fire, and the probability of sufficient combustible loading to damage critical compartment equipment before suppression. Each of these three factors was quantified in turn, and when the frequency of their combination evaluated to less than the FIVE criterion of 1E-6/yr, the compartment was screened out from further analysis. The lack of fixed combustibles in the cable spreading rooms, cable riser galleries, cable tunnel and the safety equipment building (excluding pump rooms) was a key factor in screening out compartments in Phase II.

Phase III Detailed Fire Compartment Analysis

After the simplified quantitative fire compartment analysis was performed in Phase II, the 17 remaining compartments were evaluated using more sophisticated quantitative models consistent with NUREG/CR-2300, the PRA Procedures Guide [4-3]. A comprehensive set of fire scenarios was developed for each compartment, depending on the physical configuration of the compartment and the locations of combustibles, equipment, and cables. Computer models were developed for fire growth and suppression using COMPBRN IIIE, which accounts for software deficiencies identified in NUREG/CR-5088, the Fire Risk Scoping Study [4-5]. The internal events IPE models and data were modified to reflect the fire impacts, and to include potential recovery actions. Again, those compartments with total frequency less than the 1E-6/yr criterion were screened out from further evaluation. Table 4.8-1 provides a list of the 10 compartments with calculated CDF greater than 1E-6/yr. These fire compartments are the penetration areas (2-PE-45-3A, 2-PE-63-3B, and 3-PE-63-3B), switchgear rooms 2B and 2A (2-AC-50-35 and 2-AC-50-40), switchgear room (2-AC-85-71), relay room (2-AC-9-17), turbine building (2-TB-148), and the diesel generator rooms (2-DG-30-155, 2-DG-30-158). The total CDF from these 10 compartments is 1.6E-5/yr.

TABLE 4.8-1
FIRE COMPARTMENTS NOT SCREENED BY FIVE PHASE III

FIRE COMPARTMENT	DESCRIPTION	APPROACH TO CDF REDUCTION	FINAL CDF (NOTE 1)
2-AC-50-40	Switchgear Room	Fault tree modeling	3.3E-6
2-AC-50-35	Switchgear Room	Fault tree modeling	2.9E-6
2-TB-148	Turbine Building	Fault tree modeling	2.2E-6
2-PE-63-3B	Electrical Penetration	COMBRN fire modeling	1.7E-6
3-PE-63-3B	Electrical Penetration	COMPBRN fire modeling	1.6E-6
2-PE-45-3A	Electrical Penetration	COMPBRN fire modeling	1.0E-6
2-AC-85-71	Switchgear Room	COMPBRN fire modeling	0.94E-6
2-DG-30-158	Diesel Generator Room	Fault tree modeling, Administrative change to AOI SO23-13-21,"Fire"	0.93E-6
2-DG-30-155	Diesel Generator Room	Fault tree modeling, Administrative change to AOI SO23-13-21,"Fire"	0.93E-6
2-AC-9-17	Relay Room	COMPBRN fire modeling	0.92E-6
Total CDF of Fire Compartments Not Screened By Phase III			1.6E-5/yr

Note 1: This total CDF is derived from a cutset culling limit consistent with that of the internal IPE (1.E-9/yr). CDF derived from this more rigorous calculation is slightly lower than the relatively conservative FIVE Phase III calculation shown in Table 4.4-17.

Containment Performance Analysis

Those compartments that did not screen out were evaluated to determine any unique containment performance issues, particularly with respect to the potential for containment bypass or early, large releases to the environment. Using the expanded Level 2 containment analysis from the internal events IPE, the dominant scenarios from the Phase III analysis were evaluated with the SONGS containment event tree and logic models. Table 4.8-2 provides a summary of the results and demonstrates that there are no potential vulnerabilities or unique fire-induced failures of containment or containment systems.

TABLE 4.8-2
RELEASE CATEGORY AND PROBABILITY OF FIRE IPEEE

RELEASE CATEGORY	RELEASE CATEGORY DEFINITION	RELEASE FREQUENCY (PER YEAR)	PERCENT OF TOTAL FIRE CORE DAMAGE FREQUENCY
S	Success, no containment failure within 48 hours, < 0.1% volatiles released	1.4E-5	82%
L	Late containment failure, up to 1% volatiles released	2.4E-6	15%
B	Containment bypassed, < 0.1% volatiles released	2.3E-7	1%
W	Late containment failure, more than 10% volatiles released	2.0E-7	1%
G	Early/isolation failure, containment failure prior to/at vessel failure, up to 10% volatiles released	1.4E-7	1%
D	Containment bypassed, up to 10% volatiles released	0	0
T	Containment bypassed, > 10% volatiles released	0	0

Walkdown, Fire Risk Scoping Study, and Other Issues

In accordance with the IPEEE request, walkdowns were performed to ensure that the fire models and information properly represent the as-built, as-operated plant. The checklists included in the FIVE methodology were used to address the Sandia Fire Risk Scoping Study issues. Other issues, including USI A-45 Decay Heat Removal, SECY-93-143, and seismic-fire interactions were specifically identified and evaluated for the IPEEE. No potential vulnerabilities were identified.

Conclusion

A thorough and systematic evaluation of fire events for SONGS 2/3 was performed for the IPEEE, indicating a CDF of 1.6E-5/yr. Based on this analysis, several administrative/procedural changes are being made to increase the availability and use of offsite power, and reduce the potential risk from fire events at SONGS 2/3. No potential fire-related vulnerabilities or unique failures of containment integrity or performance were identified.

The following recommendations apply to fire compartments analyzed in Phase III and will be implemented by the end of Cycle 9 Outage. These procedure modifications are need to reduce the overall CDF values for areas with higher fire vulnerability.

1. For fire compartment 2-AC-30-20A, perform an administrative change to Procedure SO23- 3-2 (Shutdown from outside the control room) to allow operators to utilize offsite power in the event that the reserve auxiliary transformers are not inadvertently tripped by fire-induced damage to panel 2/3CR-63.

2. For fire compartments 2-DG-30-155 and 2-DG-30-158, perform an administrative change to Procedure SO23-13-21 (Fire) to allow operators to recover power to the 4 kV switchgear by disconnecting power to the diesel generator feeder breaker and reclosing the offsite power breaker on the switchgear.

3. Procedure enhancement for fire/seismic/internal event risk reduction. For fire compartments 2-AC-50-44, 2-AC-50-45, 2-AC-50-46, 2-AC-50-47 (Distribution Rooms), 2-AC-50-35 and 2-AC-50-40 (Switchgear Rooms), implementation of an administrative change to alarm response procedure SO23-15-60.A1 (Annunciator Panel 60A, Emergency HVAC) would allow operator to use air duct and gas driven fans to prevent room heat-up.

4.9 REFERENCES

4-1 "Fire-Induced Vulnerability Evaluation (FIVE)," EPRI TR-100370, April 1992.

4-2 "COMPBRN IIIE: An Interactive Computer Code for Fire Risk Analysis," EPRI Report NP-7282, May 1991.

4-3 "PRA Procedure Guide," NUREG/CR-2300, January 1983.

4-4 "Individual Plant Examination Report for San Onofre Nuclear Generating Station Units 2 and 3 - In Response to Generic Letter 88-20," April 1993, CDM Document No. C930614S6110.

4-5 "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," NUREG/CR-5088, January 1989.

4-6 Calculation A-92-MN-003, IPEEE-Fire: Phase I, Revision 0.

4-7 Calculation A-MN-92-004, IPEEE-Fire: Phase II, III, Revision 0.

4.8 NRC IE Notice 89-52, "Potential Fire Damper Operational Problems."

4-9 "Updated Fire Hazards Analysis (UFHA), San Onofre Nuclear Generating Station - Units 1, 2 & 3"

4.10 NUREG/CR-2258, "Fire Risk Analysis for Nuclear Power Plants," University of California Los Angeles, September 1981.

4.11 Fire Events Database for U.S. Nuclear Power Plants, SAIC Report NSAC 178L, June 1992.

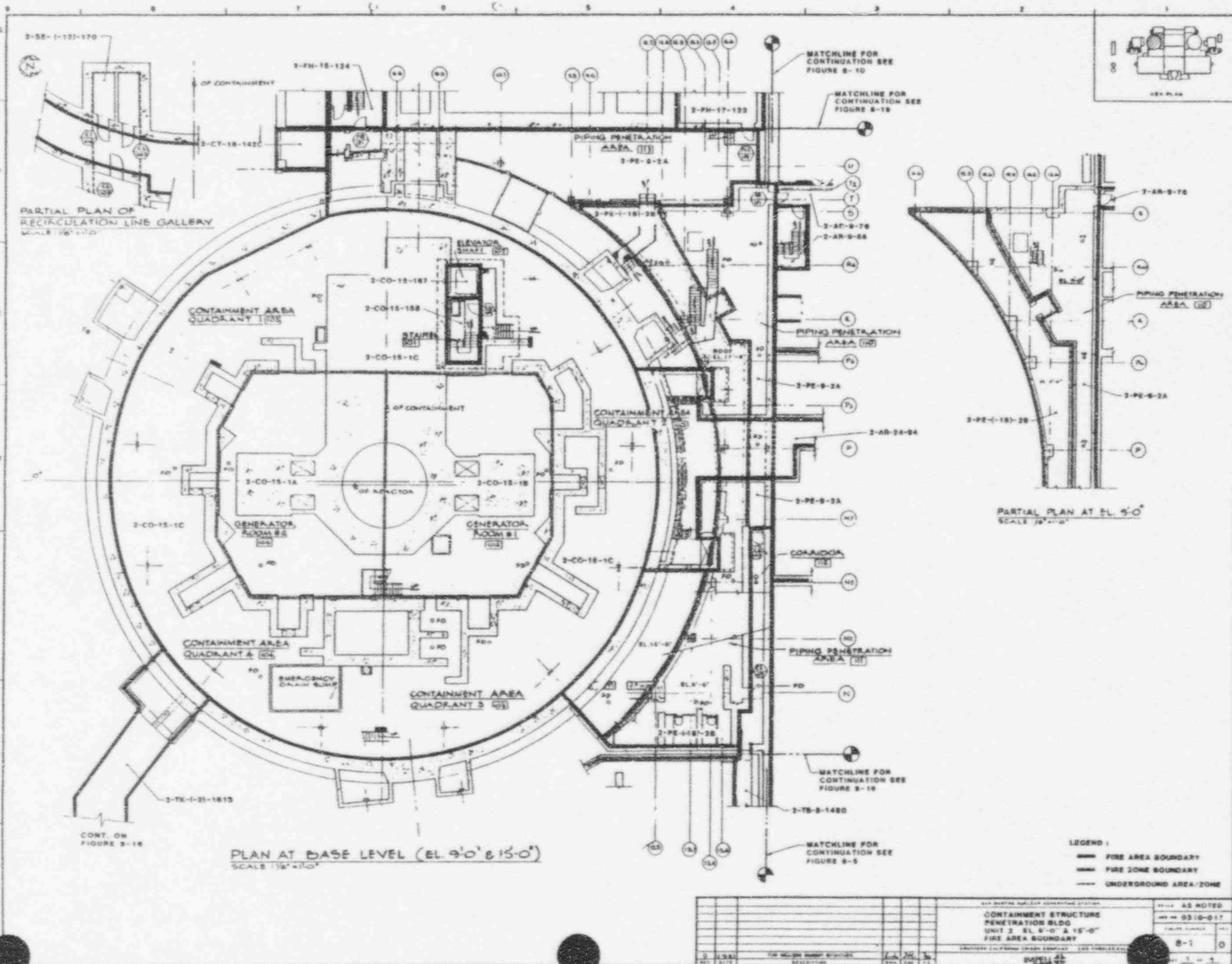
4.12 "NUCAP+ User's Manual," Version 2.0, NUS Report 5282, Revision 2, September 1992.

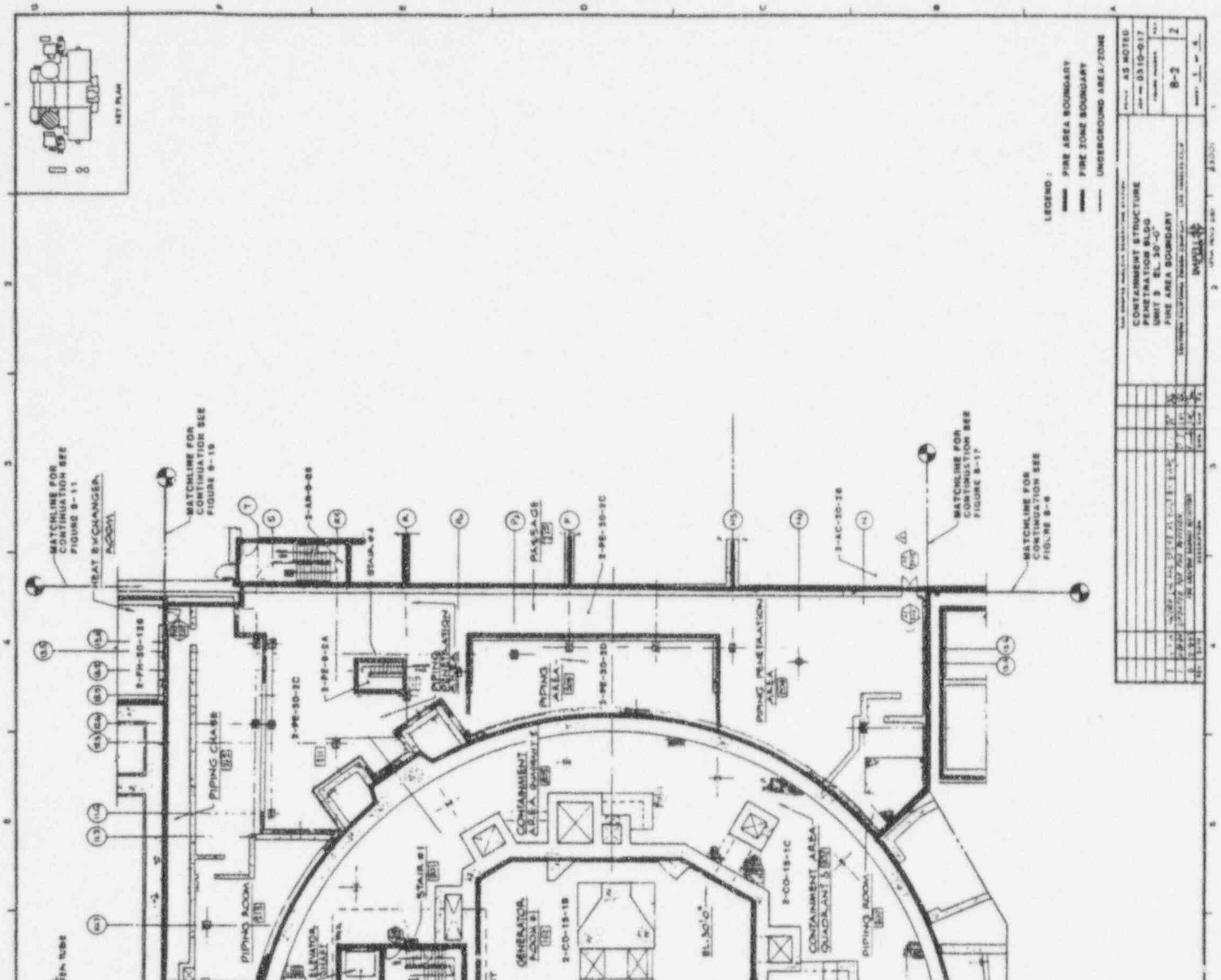
4.13 NRC IE Notice 83-41, "Actuation of Fire Suppression Systems Causing Inoperability of Safety Related Equipment."

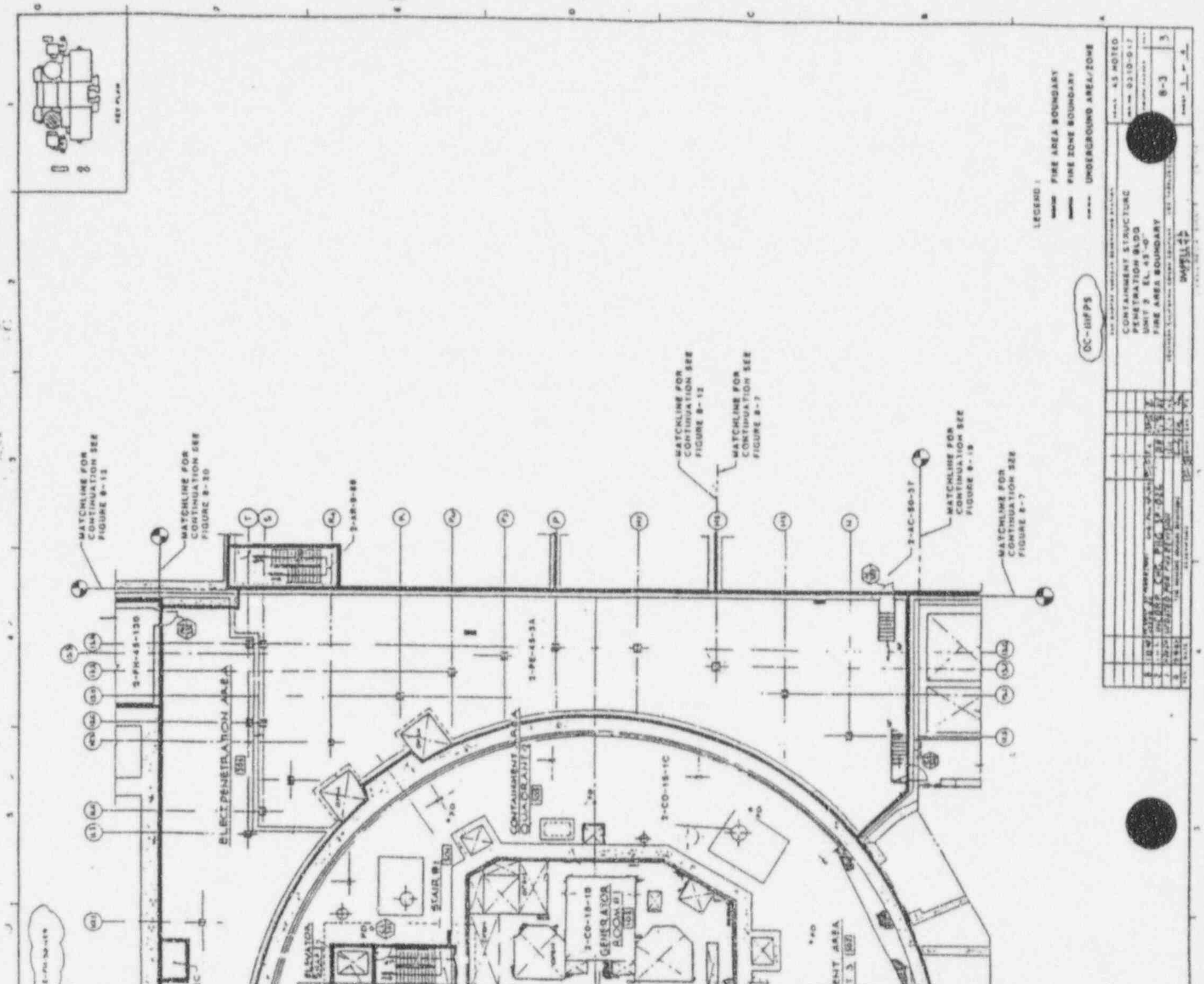
4.10 FIRE APPENDICES

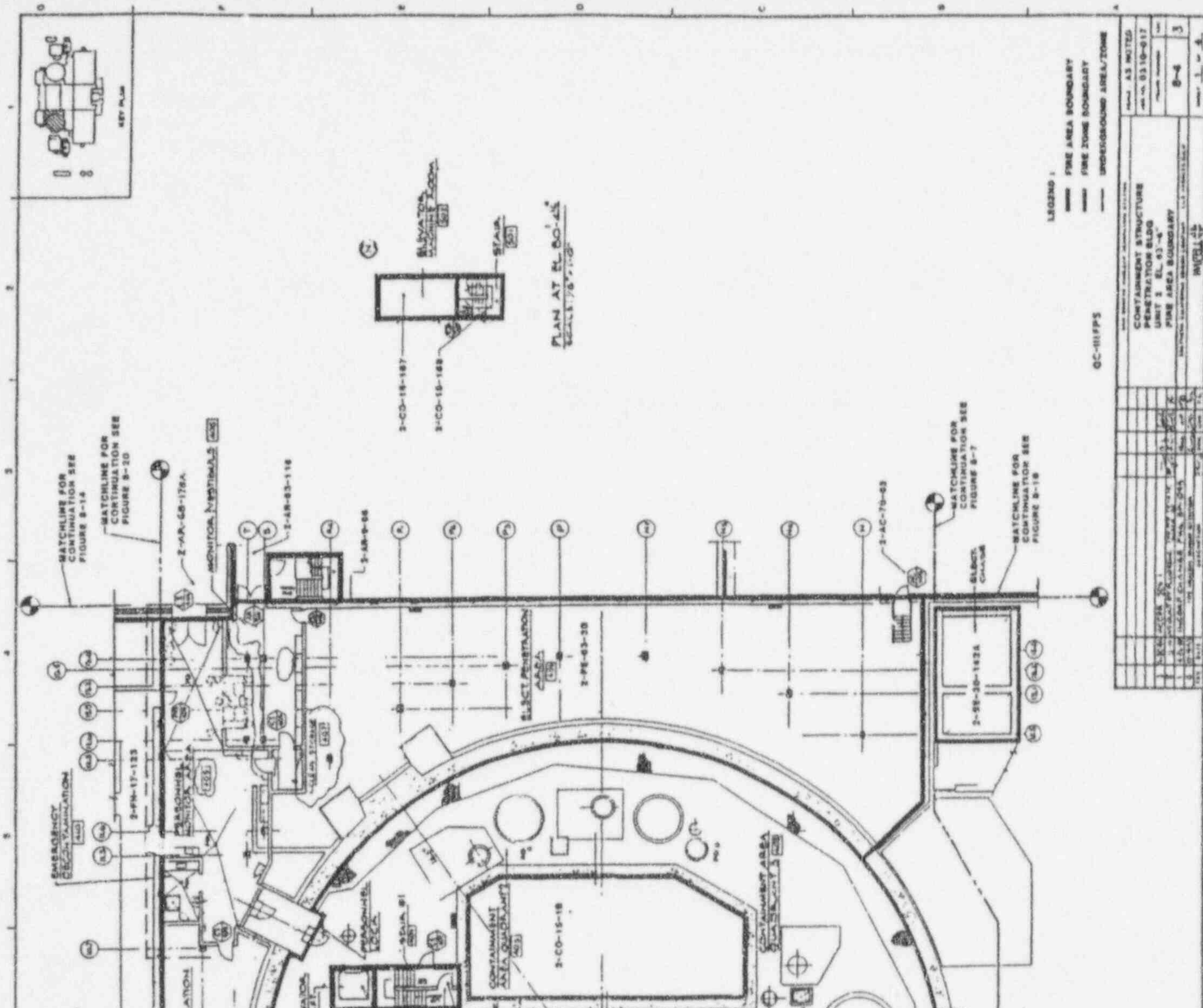
SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 & 3

FIRE AREA/COMPARTMENT DRAWINGS

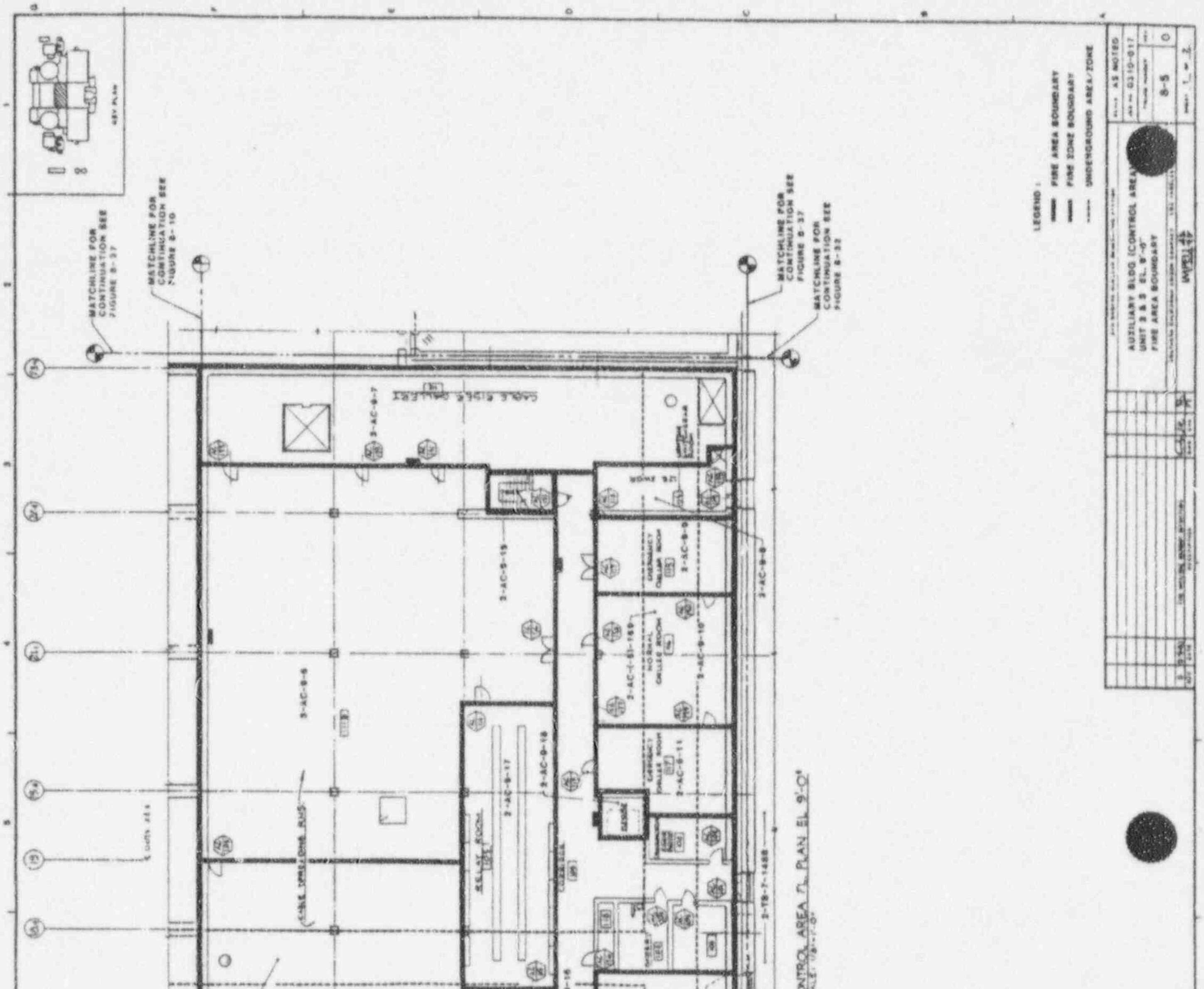

PARTIAL PLAN OF RECIRCULATION LINE GALLERY
2-SE-(-12)-170
2-CT-18-142C
2-FH-15-124
2-FH-17-122
PIPING PENETRATION AREA
2-PE-9-2A
MATCHLINE FOR CONTINUATION SEE FIGURE 8-10
MATCHLINE FOR CONTINUATION SEE FIGURE 8-19
KEY PLAN
ELEVATOR SHAFT
2-CO-15-167
2-CO-15-158
STAIRS
2-CO-15-1C
CONTAINMENT AREA QUADRANT 1
CONTAINMENT AREA QUADRANT 2
CONTAINMENT AREA QUADRANT 3
CONTAINMENT AREA QUADRANT 4
2-CO-15-1A
2-CO-15-1B
GENERATOR ROOM #2
GENERATOR ROOM #1
EMERGENCY DRAIN SUMP
2-AR-9-66
2-AR-24-94
2-PE-(-18)-28
7-AR-9-76
CORRIDOR
2-TK-(-21)-1615
CONT. ON FIGURE 8-16
MATCHLINE FOR CONTINUATION SEE FIGURE 8-16
2-TB-8-1480
MATCHLINE FOR CONTINUATION SEE FIGURE 8-5
PARTIAL PLAN AT EL. 9'-0"
PLAN AT BASE LEVEL (EL. 9'-0" & 15'-0")
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
CONTAINMENT STRUCTURE
PENETRATION BLDG
UNIT 2 EL. 9'-0" & 15'-0"
FIRE AREA BOUNDARY
AS NOTED
8-1
0


LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
CONTAINMENT STRUCTURE
PENETRATION BLDG
UNIT 3 EL 30'-0"
FIRE AREA BOUNDARY
B-2
CONTAINMENT AREA QUADRANT 1
CONTAINMENT AREA QUADRANT 4
GENERATOR ROOM #1
GENERATOR ROOM #2
REFUELING CANAL
PIPING CHASE
PIPING ROOM
PASSAGE
PIPING PENETRATION AREA
ELEVATOR
STAIR #1
PLAN AT EL 30'-0"

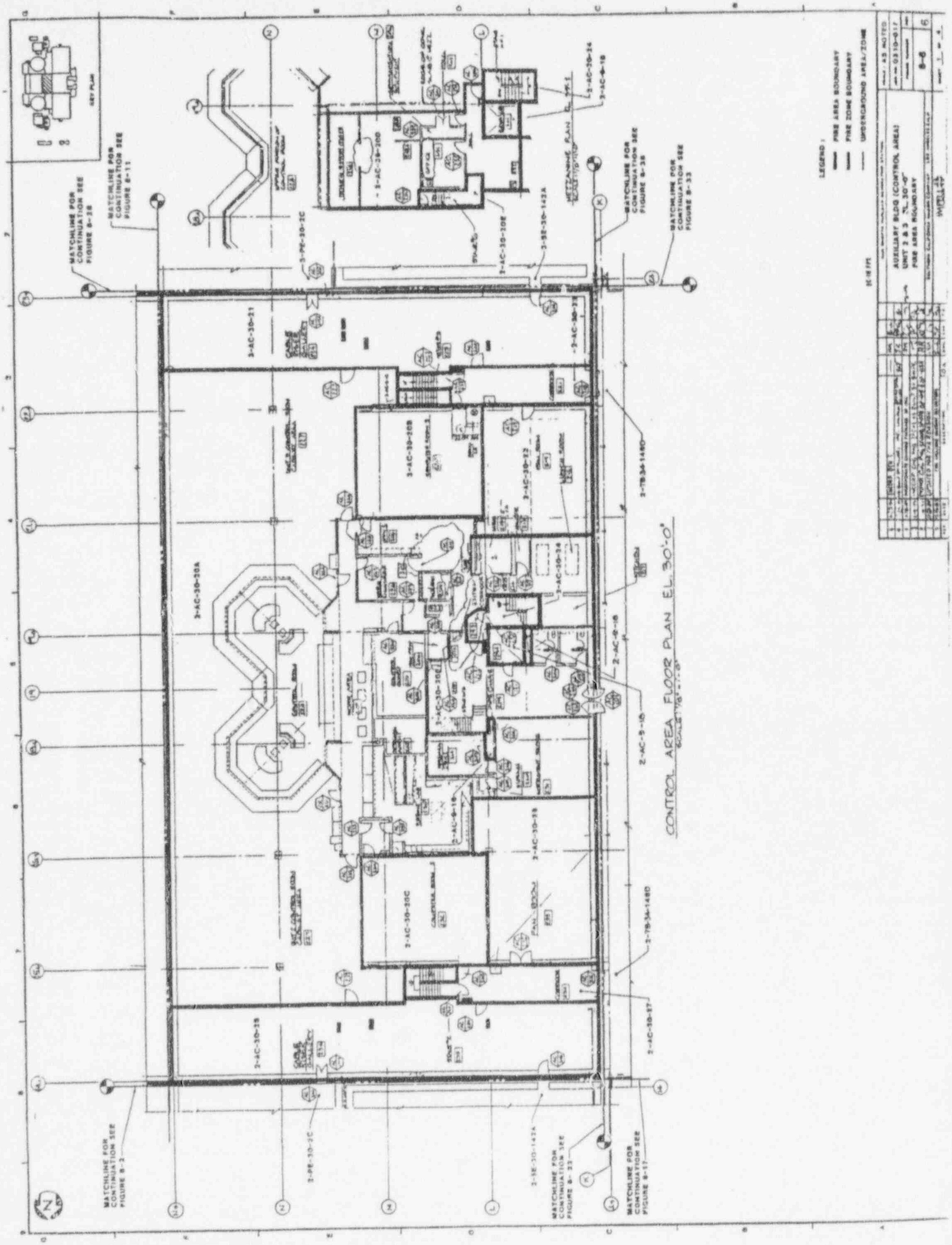

MATCHLINE FOR CONTINUATION SEE FIGURE 8-12
MATCHLINE FOR CONTINUATION SEE FIGURE 8-20
MATCHLINE FOR CONTINUATION SEE FIGURE 8-7
MATCHLINE FOR CONTINUATION SEE FIGURE 8-19
A/C ROOM
ELECT. CABLE RISER SHAFT
2-CT-16-142C
2-FH-45-131
2-PE-30-2C
2-FH-45-130
ELECT. PENETRATION AREA
2-AR-8-88
CONTAINMENT AREA QUADRANT
REFUELING CANAL
2-CO-15-187
2-CO-15-188
2-CO-15-1C
ELEVATOR SHAFT
STAIR
CONTAINMENT AREA QUADRANT 2
2-PE-45-3A
2-CO-15-1A
GENERATOR ROOM #2
2-CO-15-1B
GENERATOR ROOM #1
CONTAINMENT AREA QUADRANT 4
CONTAINMENT AREA QUADRANT 3
2-AC-50-37
PLAN AT EL. 45'-0"
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
OC-111FPS
CONTAINMENT STRUCTURE
PENETRATION BLDG
UNIT 2 EL. 45'-0"
FIRE AREA BOUNDARY
AS NOTED
8-3
KEY PLAN

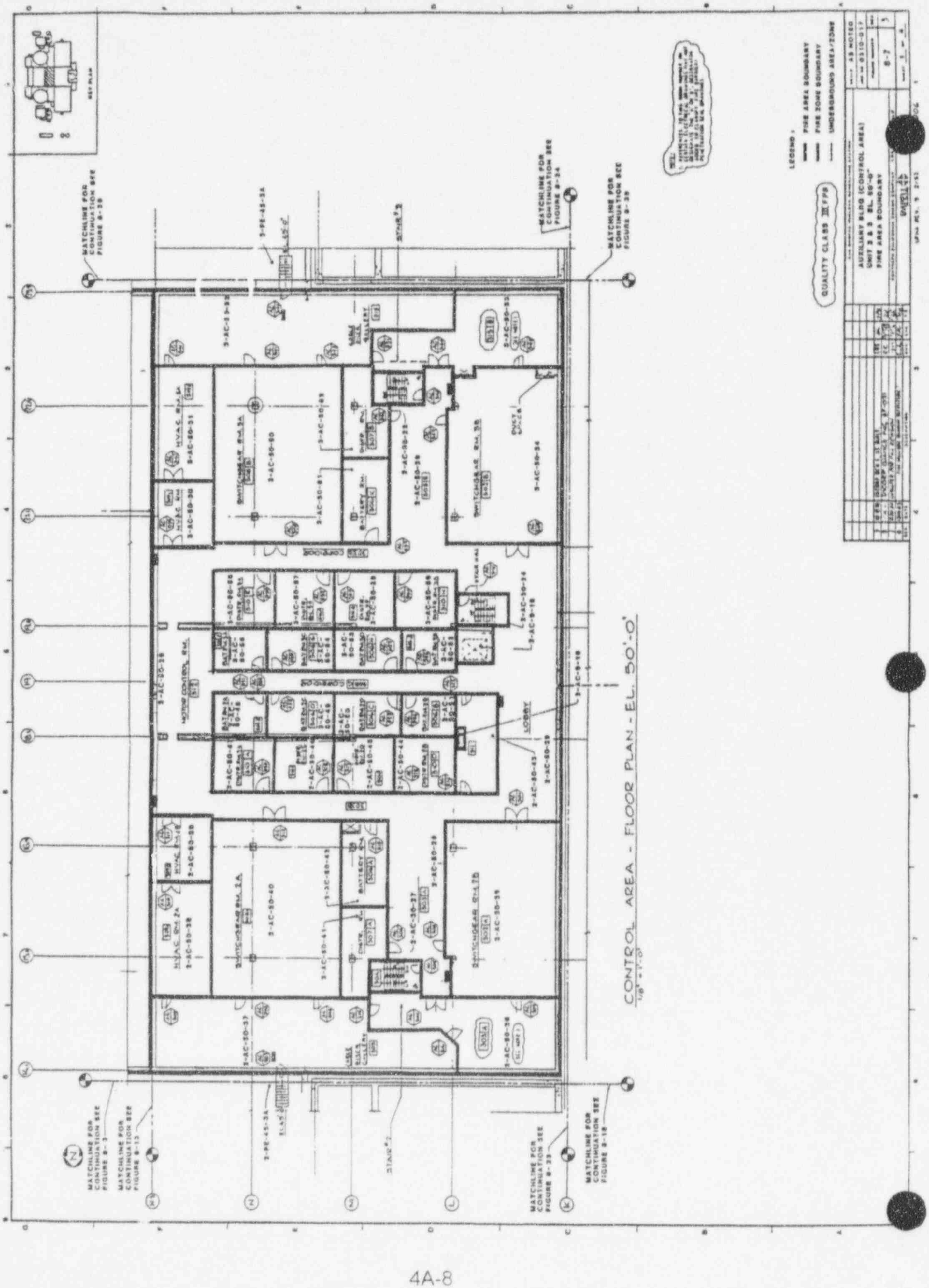


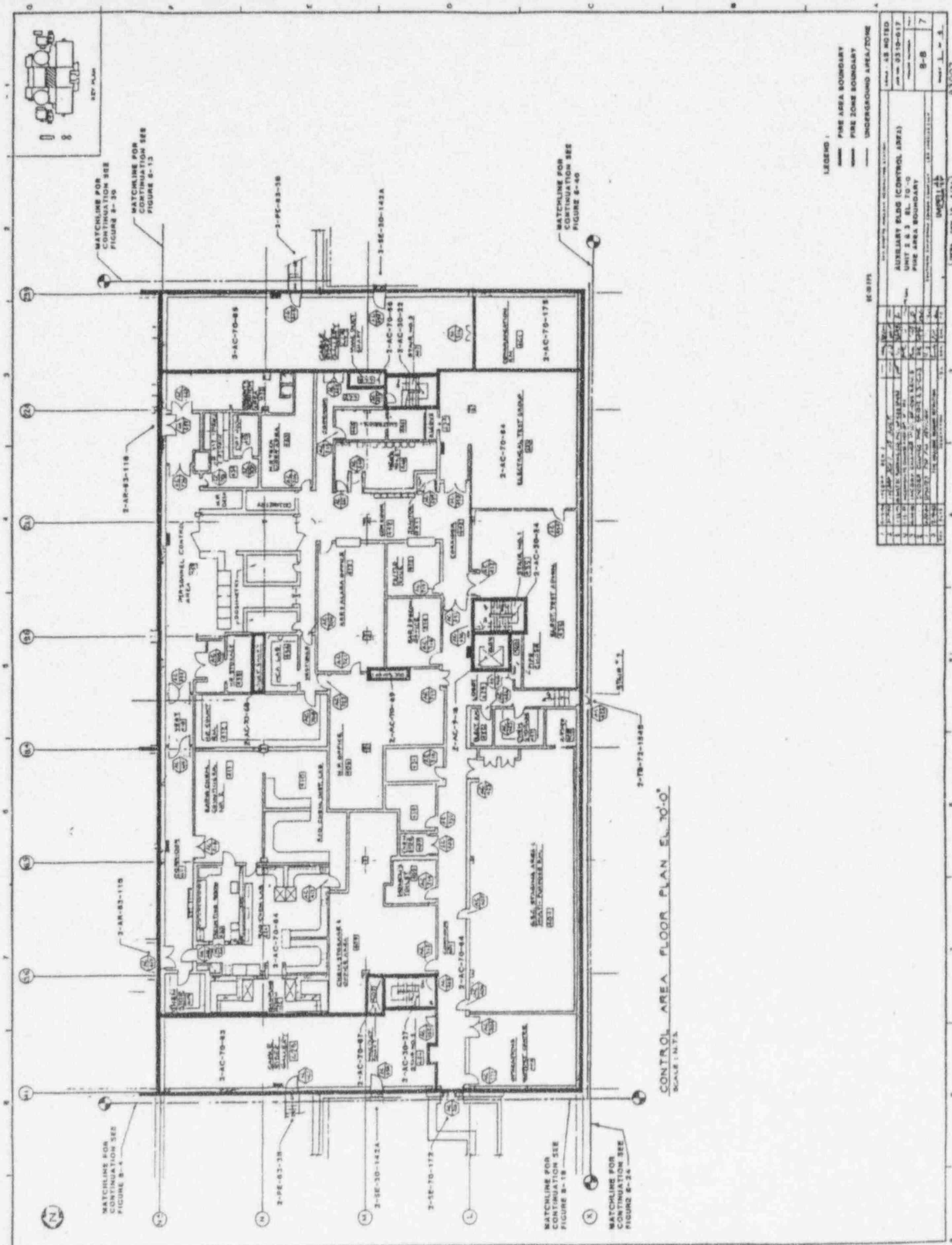
LEGEND :
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
CONTAINMENT STRUCTURE
PENETRATION BLDG
UNIT 2 EL 63'-6"
FIRE AREA BOUNDARY
AS NOTED
8-4
GC-111FPS
WATCHLINE FOR CONTINUATION SEE FIGURE 8-19
PLAN AT EL 63'-6"
KEY PLAN
2-CO-63-10
OPERATING FLOOR
2-CO-15-1A
2-CO-15-1B
CONTAINMENT AREA

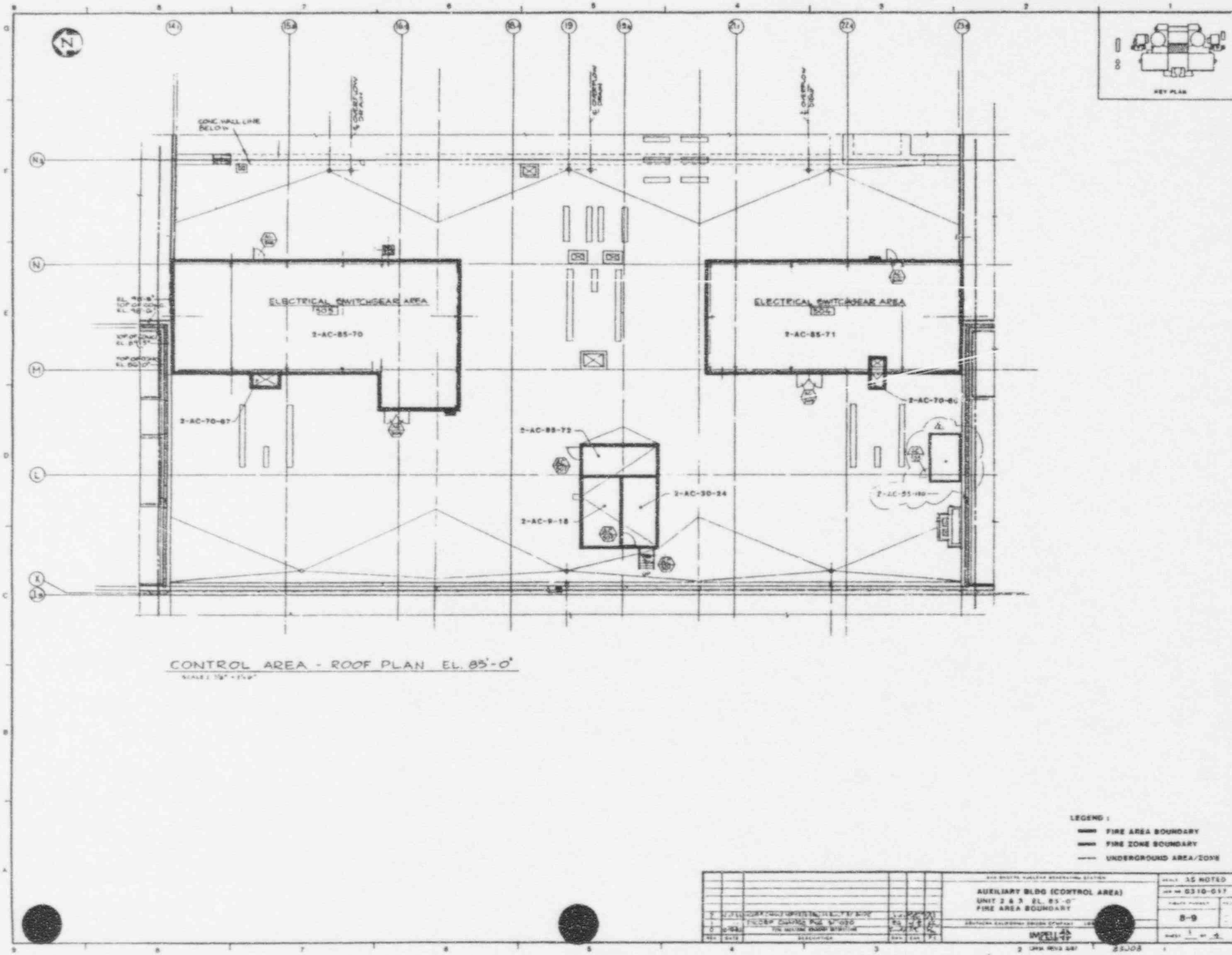


CONTROL AREA FL. PLAN EL 9'-0"
RELAY ROOM
CABLE RISER GALLERY
MATCHLINE FOR CONTINUATION SEE FIGURE 8-37
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (CONTROL AREA)
FIRE AREA BOUNDARY

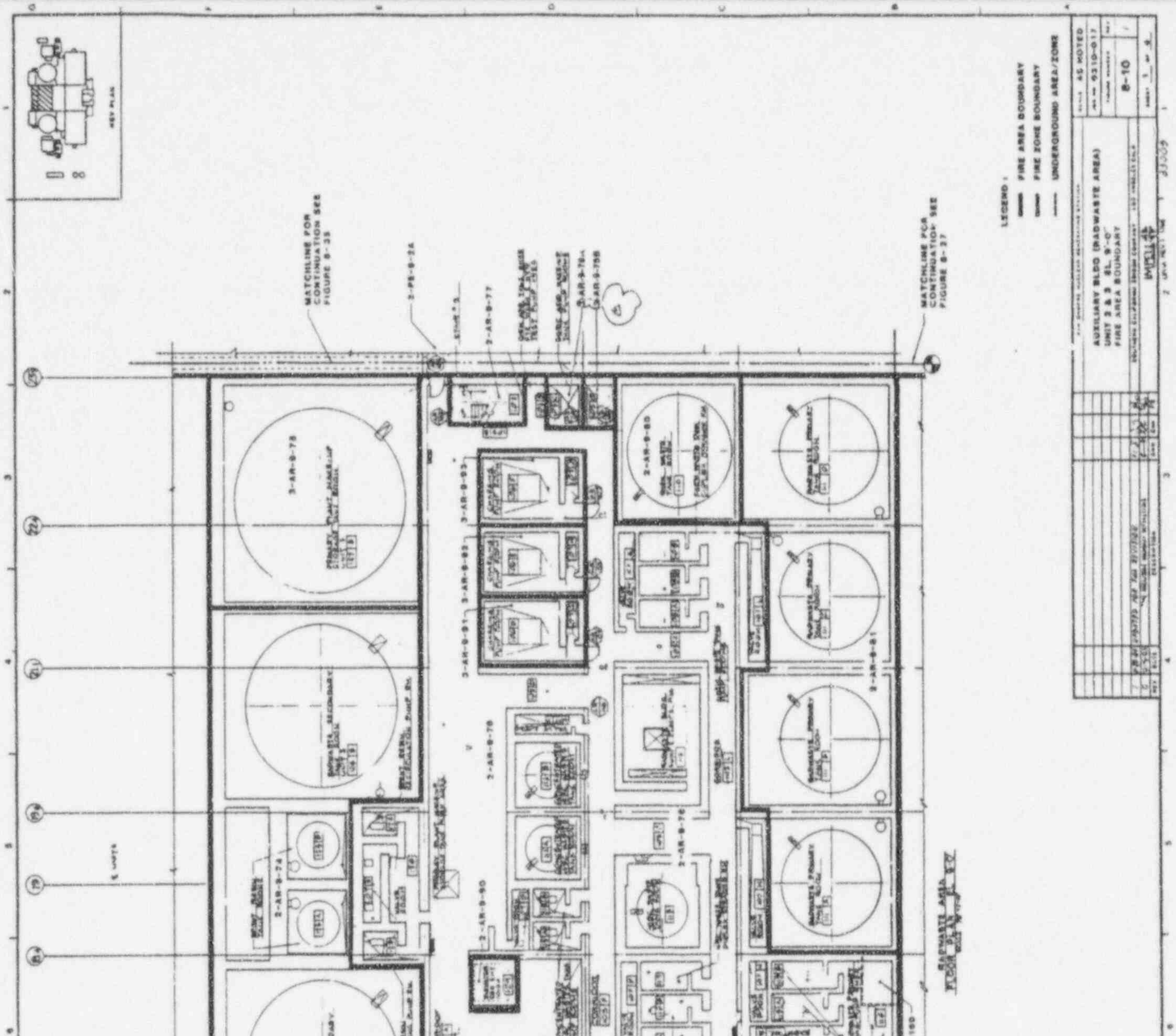

CONTROL AREA FLOOR PLAN EL. 30'-0"
MEZZANINE PLAN EL. 39'-0"
MATCHLINE FOR CONTINUATION SEE FIGURE 8-2
MATCHLINE FOR CONTINUATION SEE FIGURE 8-33
MATCHLINE FOR CONTINUATION SEE FIGURE 8-17
MATCHLINE FOR CONTINUATION SEE FIGURE 8-11
MATCHLINE FOR CONTINUATION SEE FIGURE 8-28
CONTROL ROOM
COMPUTER ROOM
FAN ROOM
CABLE SPREADING GALLERY
2-AC-30-20A
2-AC-30-20C
2-AC-30-23
3-AC-30-21
3-AC-30-20B
3-AC-30-22
3-AC-30-24
2-RE-30-2C
3-RE-30-2C
2-SE-30-142A
3-SE-30-142A
2-TB-34-148D
3-TB-34-148D
2-AC-8-18
2-AC-5-18
3-AC-8-18
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (CONTROL AREA)
UNIT 2 & 3 EL. 30'-0"
FIRE AREA BOUNDARY
KEY PLAN

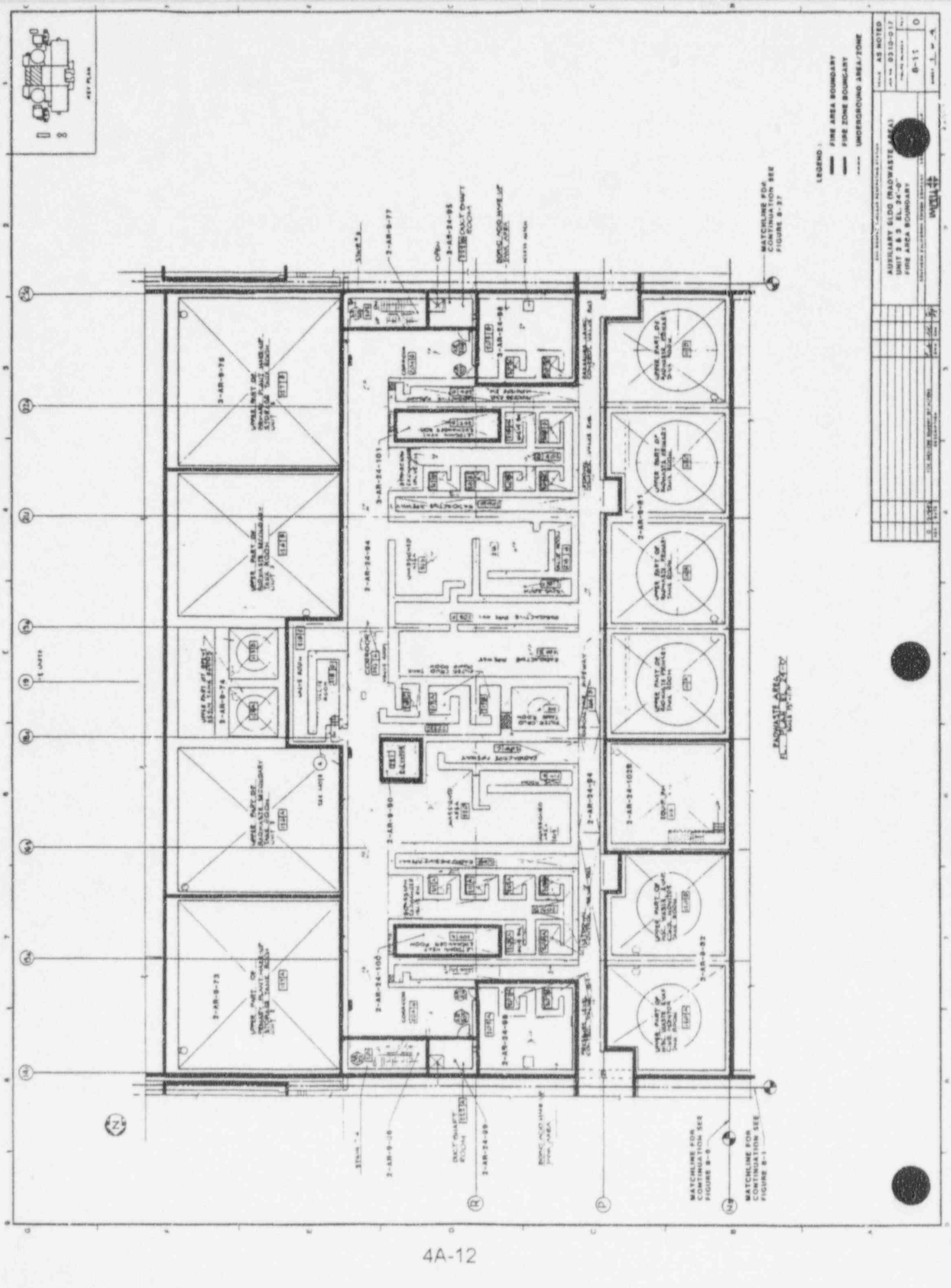

CONTROL AREA - FLOOR PLAN - EL. 50'-0"

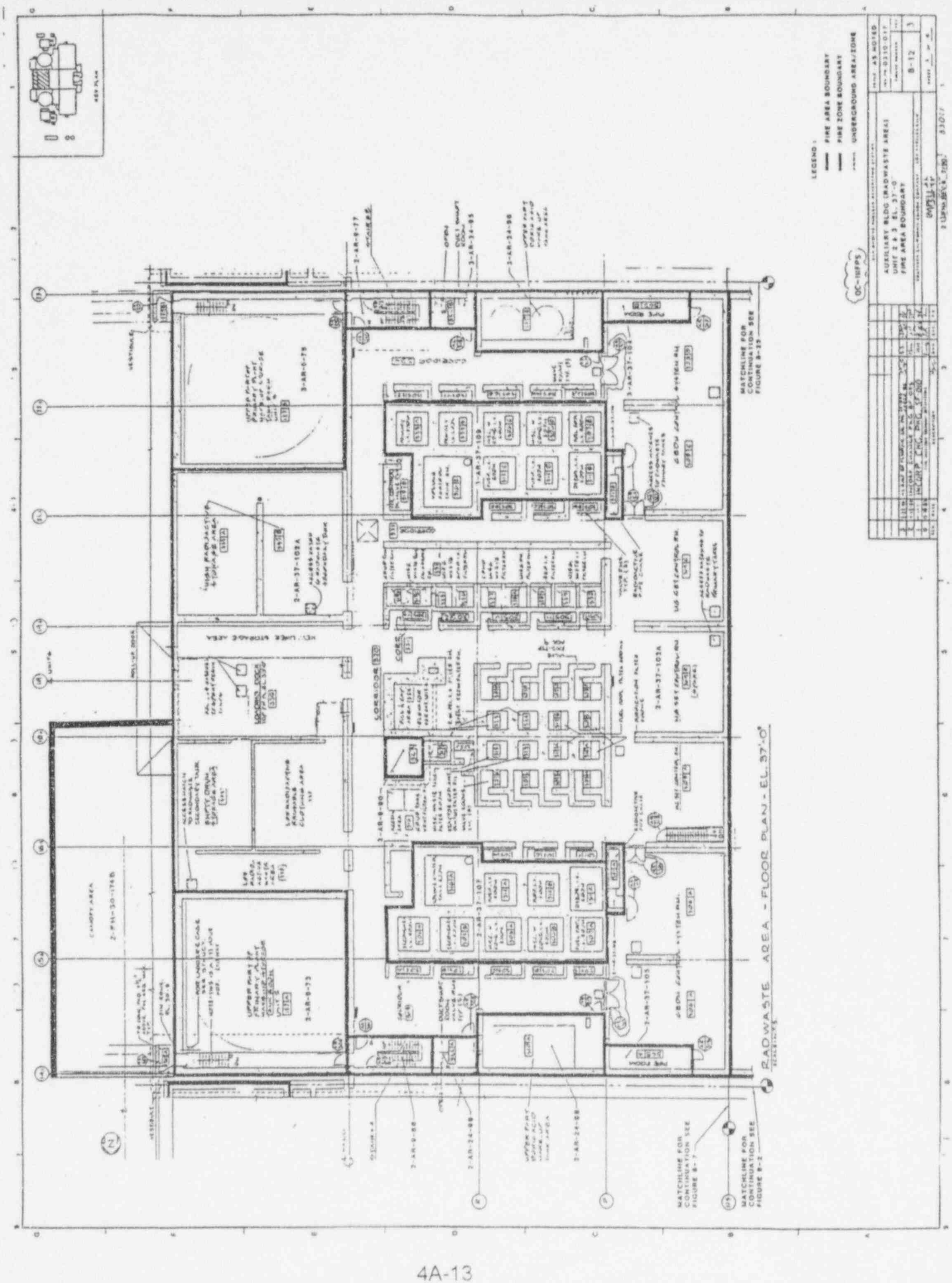

CONTROL AREA FLOOR PLAN EL. 70'-0"

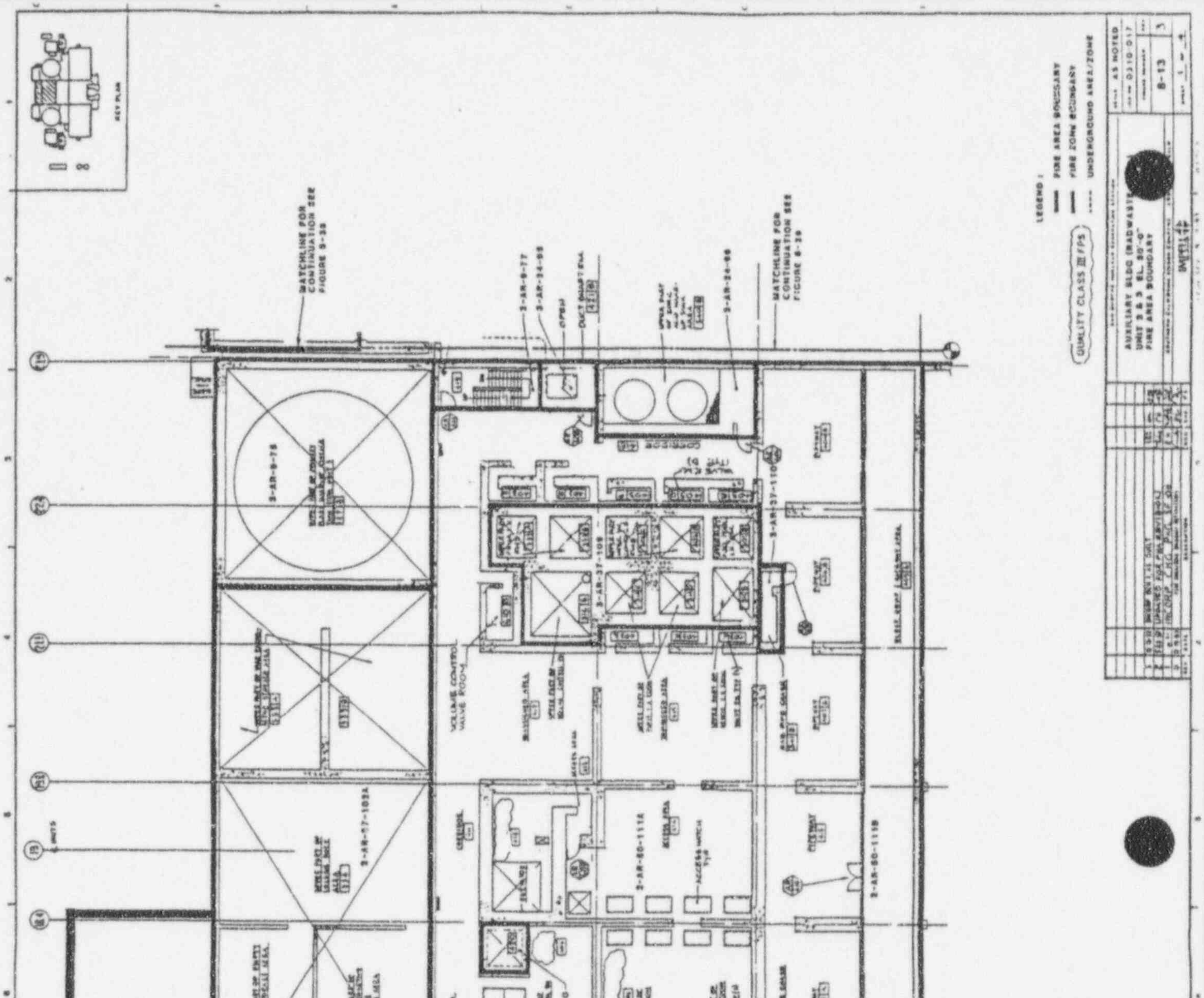


CONTROL AREA - ROOF PLAN EL. 85'-0"

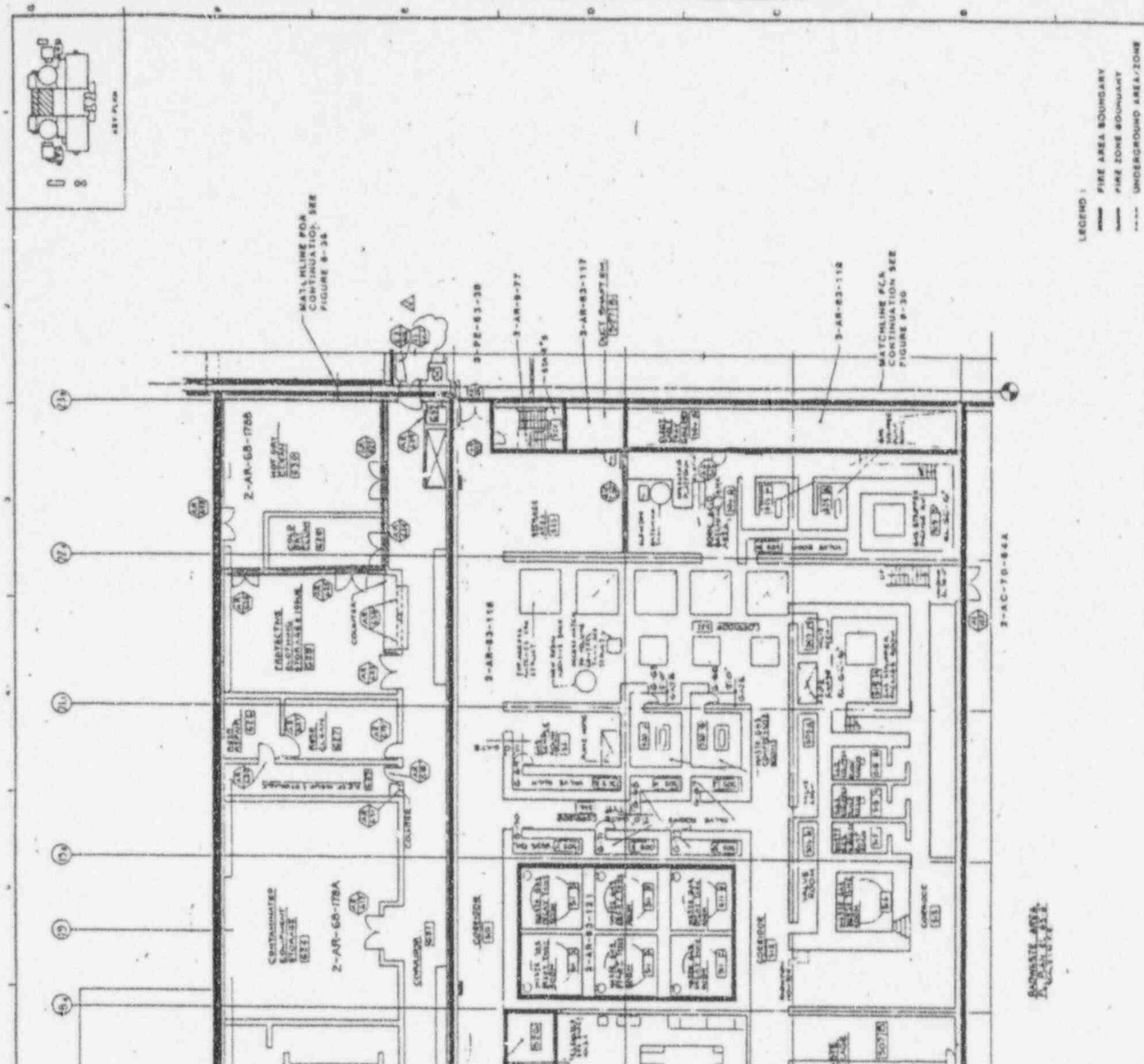

LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (RADWASTE AREA)
UNIT 2 & 3 EL 9'-0"
FIRE AREA BOUNDARY
8-10
MATCHLINE FOR CONTINUATION SEE FIGURE 8-1
MATCHLINE FOR CONTINUATION SEE FIGURE 8-5
MATCHLINE FOR CONTINUATION SEE FIGURE 8-19


LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (RADWASTE AREA)
UNIT 2 & 3 EL. 24'-0"
FIRE AREA BOUNDARY
AS NOTED
0310-017
8-11
0
RADWASTE AREA
MATCHLINE FOR CONTINUATION SEE FIGURE 8-37
MATCHLINE FOR CONTINUATION SEE FIGURE 8-5
MATCHLINE FOR CONTINUATION SEE FIGURE 8-1
KEY PLAN

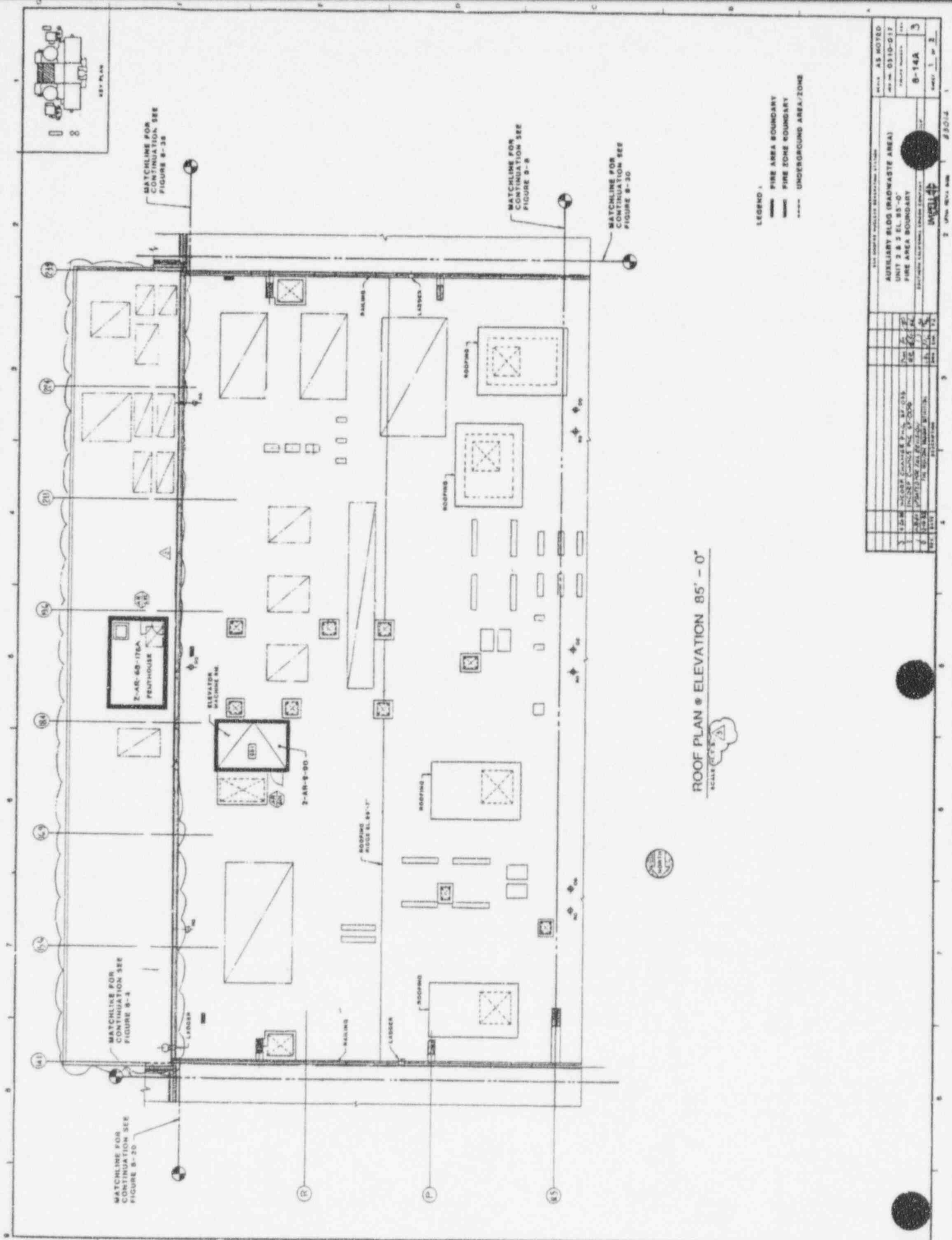

RADWASTE AREA - FLOOR PLAN - EL. 37'-0"
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (RADWASTE AREA)
UNIT 2 & 3 EL. 37'-0"
FIRE AREA BOUNDARY
MATCHLINE FOR CONTINUATION SEE FIGURE 8-2
KEY PLAN
CORRIDOR
CANOPY AREA

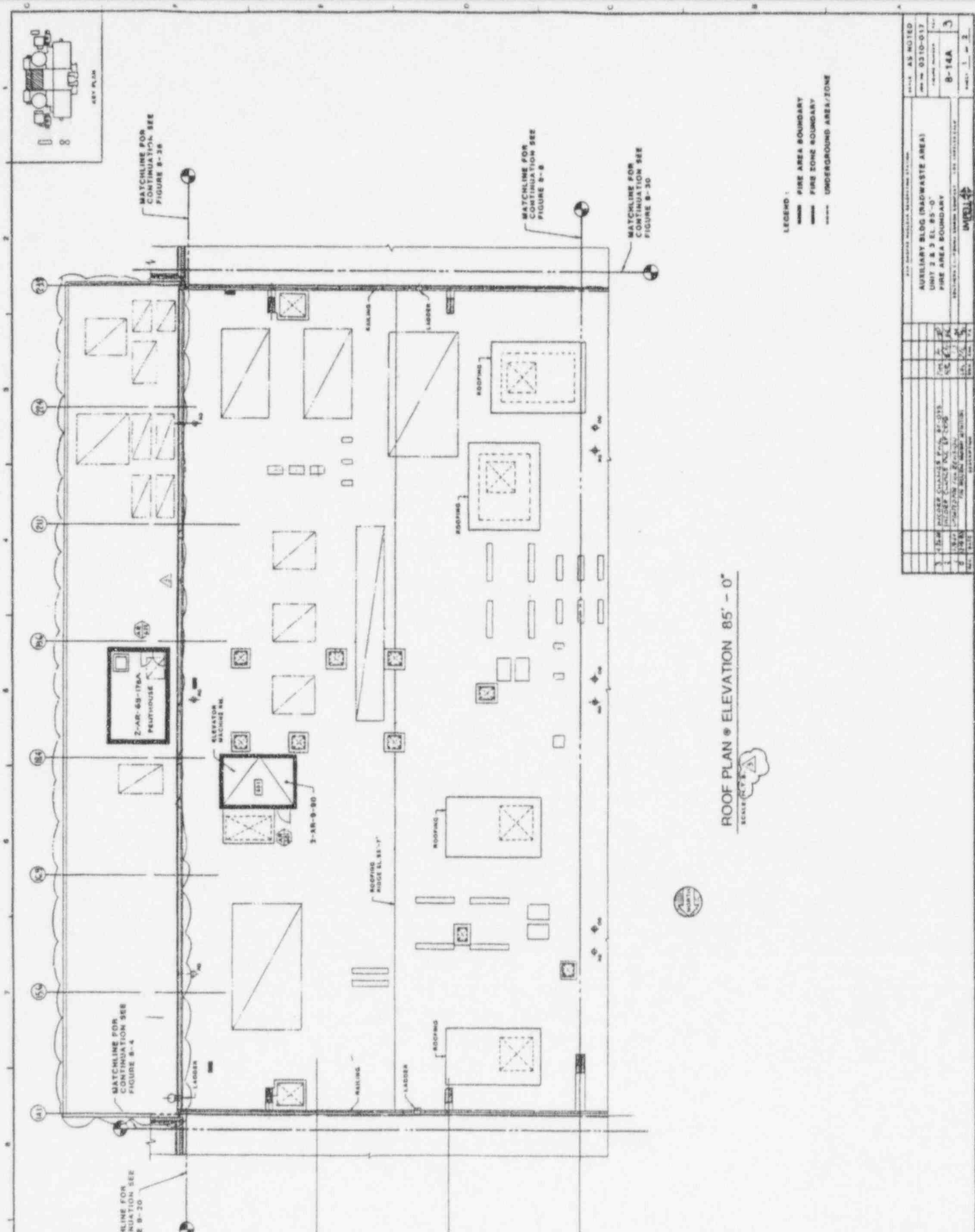


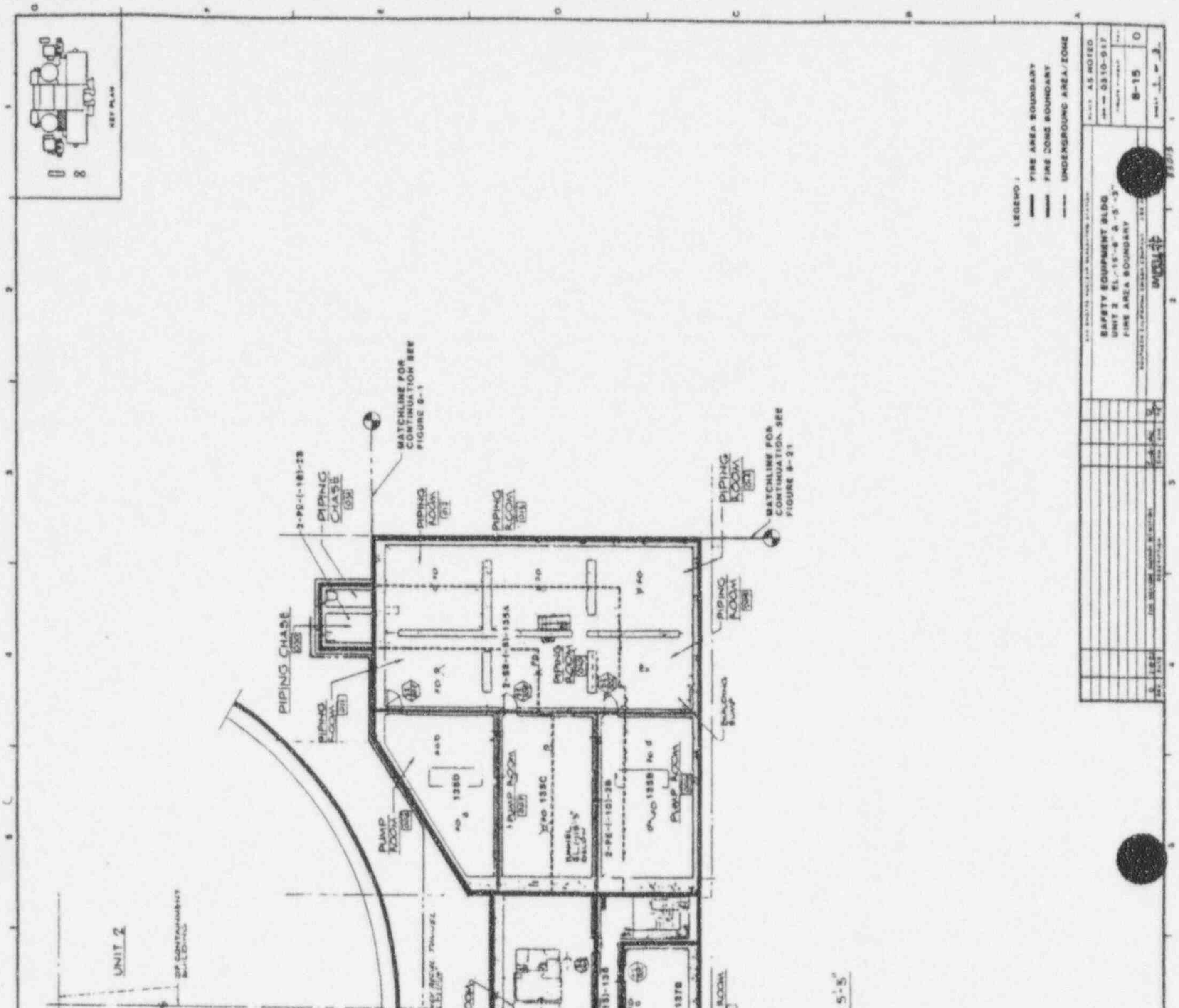
RADWASTE AREA
PLAN - EL 50'-0"
VOLUME CONTROL VALVE ROOM
CANOPY AREA
MATCHLINE FOR CONTINUATION SEE FIGURE 8-20
MATCHLINE FOR CONTINUATION SEE FIGURE 8-7
MATCHLINE FOR CONTINUATION SEE FIGURE 8-3
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
QUALITY CLASS III FPS
KEY PLAN



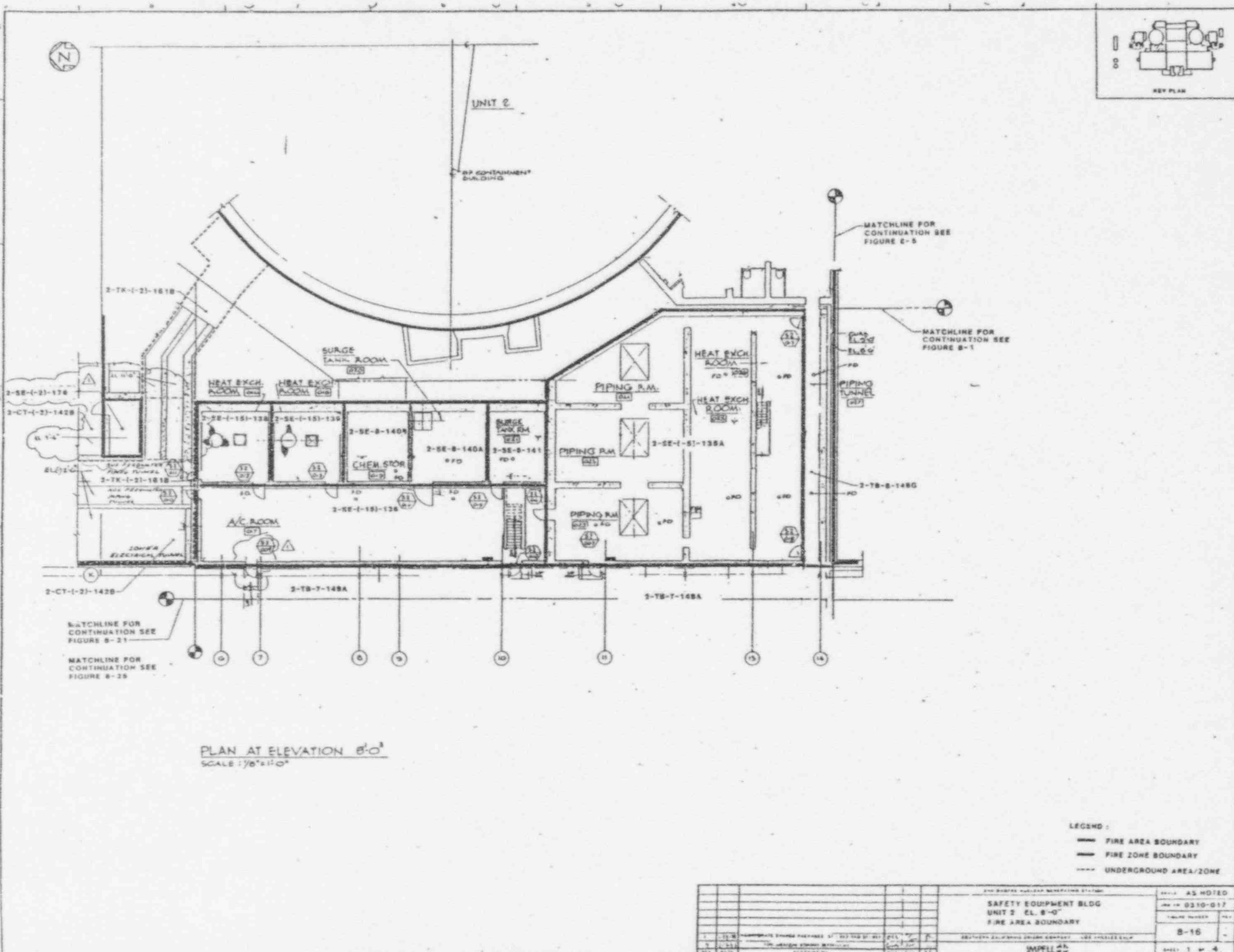
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (RADWASTE AREAS)
UNIT 2 & 3 EL 63'-6"
FIRE AREA BOUNDARY
8-14

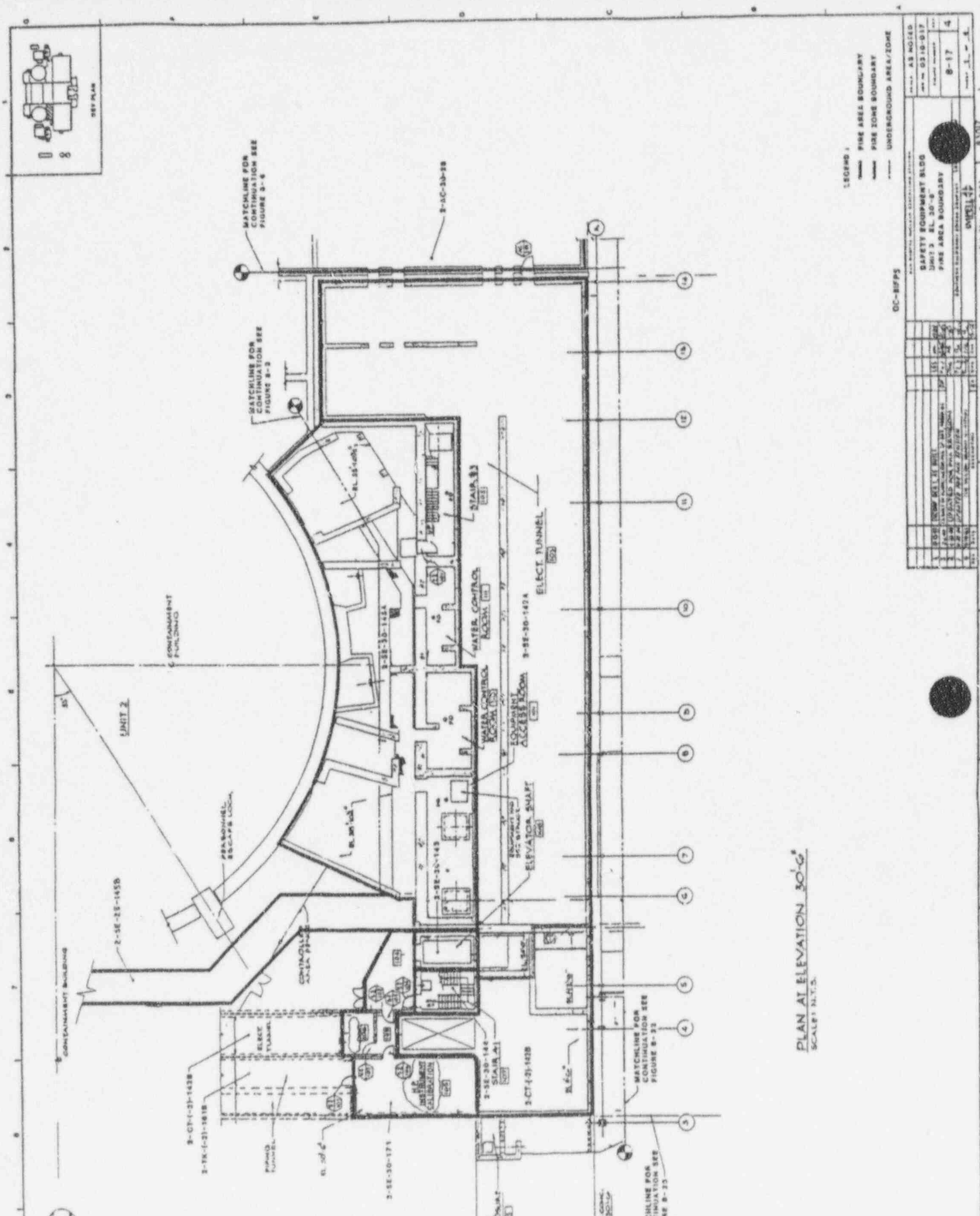

ROOF PLAN • ELEVATION 85' - 0"
PENTHOUSE
ELEVATOR MACHINE RM.
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (RADWASTE AREA)
UNIT 2 & 3 EL. 85'-0"
FIRE AREA BOUNDARY

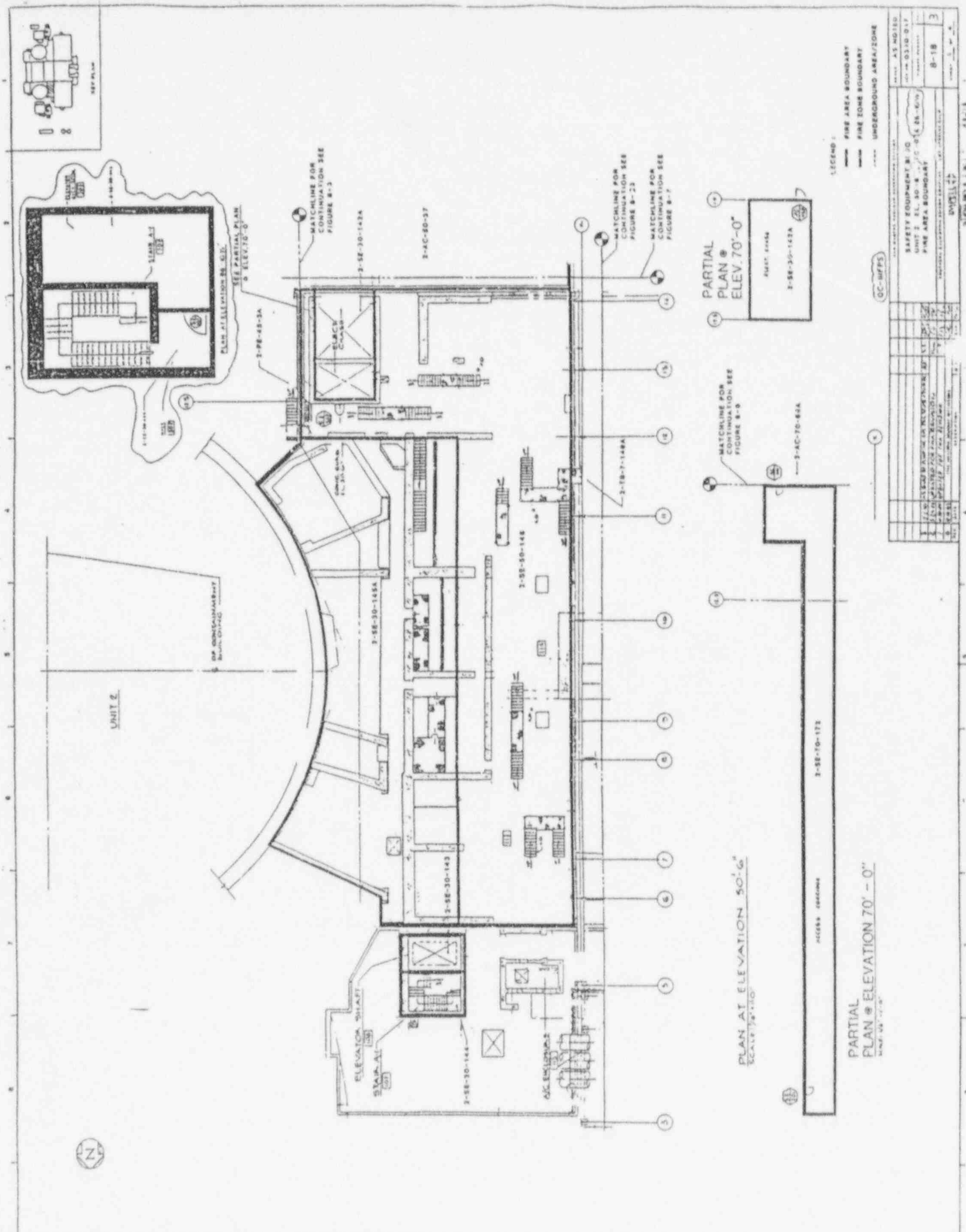

ROOF PLAN @ ELEVATION 85' - 0"
MATCHLINE FOR CONTINUATION SEE FIGURE 8-26
MATCHLINE FOR CONTINUATION SEE FIGURE 8-8
MATCHLINE FOR CONTINUATION SEE FIGURE 8-30
MATCHLINE FOR CONTINUATION SEE FIGURE 8-4
MATCHLINE FOR CONTINUATION SEE FIGURE 8-20
KEY PLAN
Z-AR-65-178A PENTHOUSE
ELEVATOR MACHINE RM.
3-AR-8-90
RAILING
LADDER
ROOFING
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
AUXILIARY BLDG (RADWASTE AREA)
UNIT 2 & 3 EL 85'-0"
FIRE AREA BOUNDARY
AS NOTED
0310-017
8-14A
3
83014

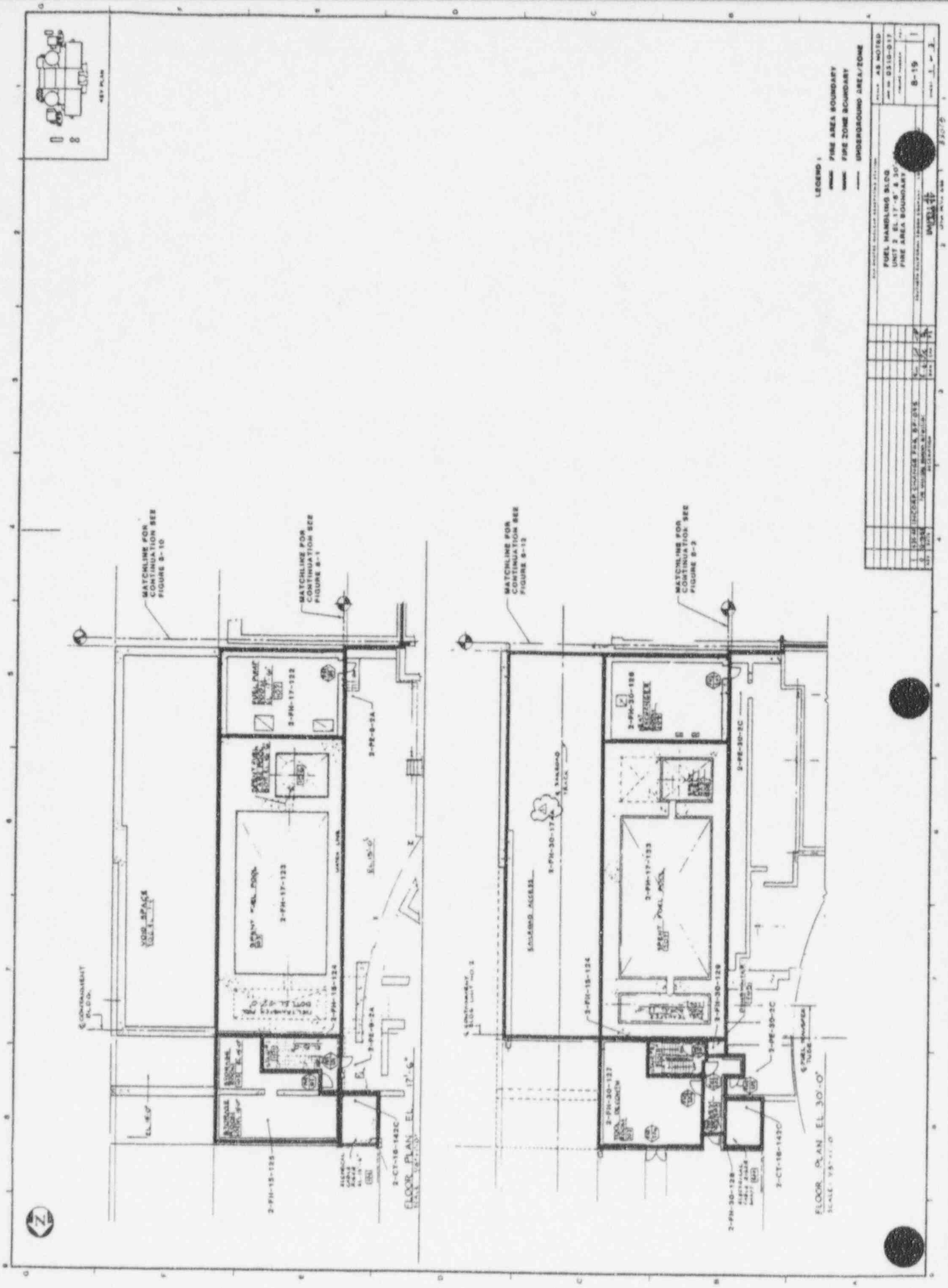


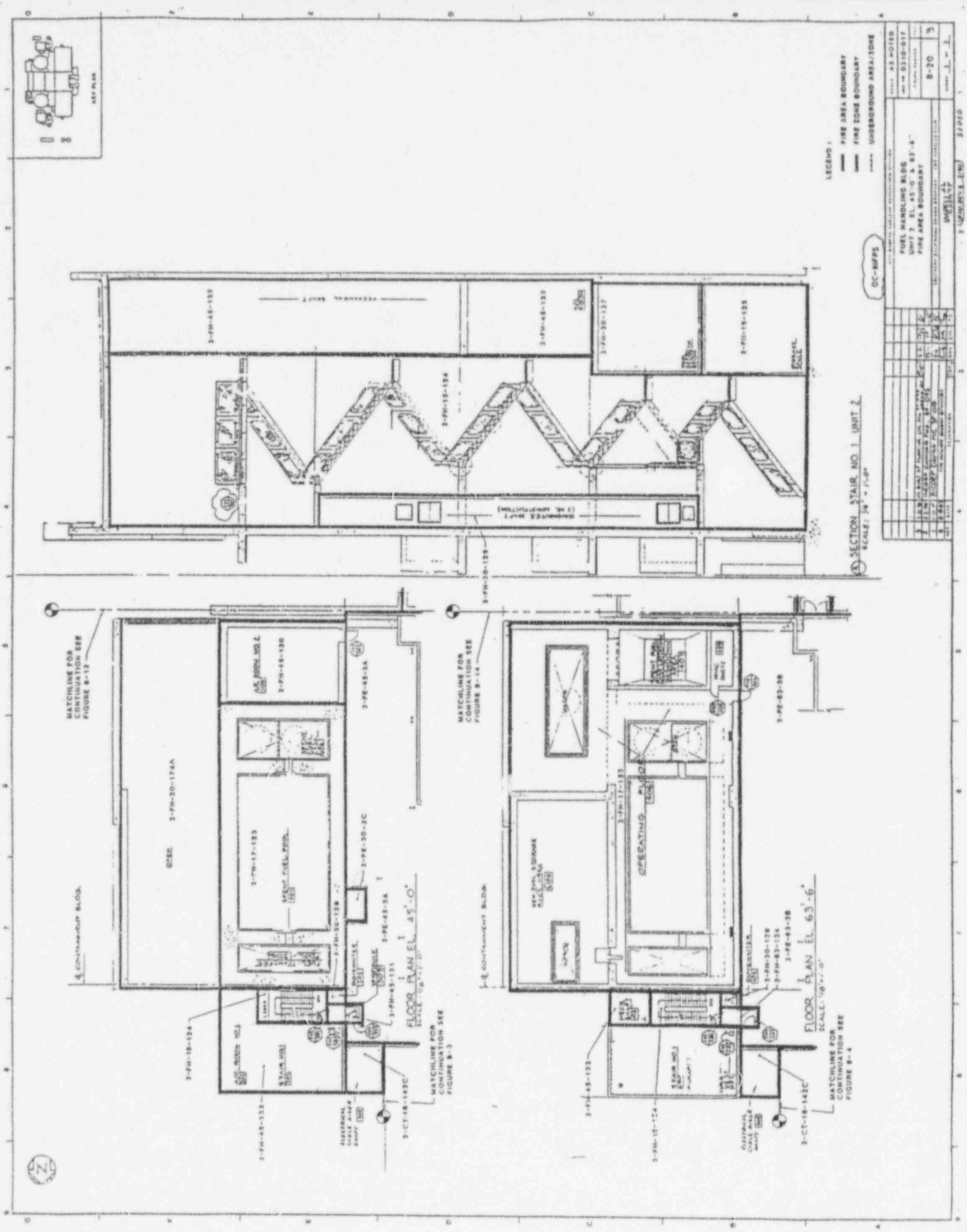
KEY PLAN
UNIT 2
PIPING CHASE
PIPING ROOM
PUMP ROOM
HEAT EXCH. ROOM
MATCHLINE FOR CONTINUATION SEE FIGURE 8-1
PLAN AT ELEVATION (-)15'-6 & (-)5'-5"
PLAN AT PLATFORM EL.(-)2'0"
ABOVE HEAT EXCH. ROOMS
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
SAFETY EQUIPMENT BLDG
FIRE AREA BOUNDARY
8-15

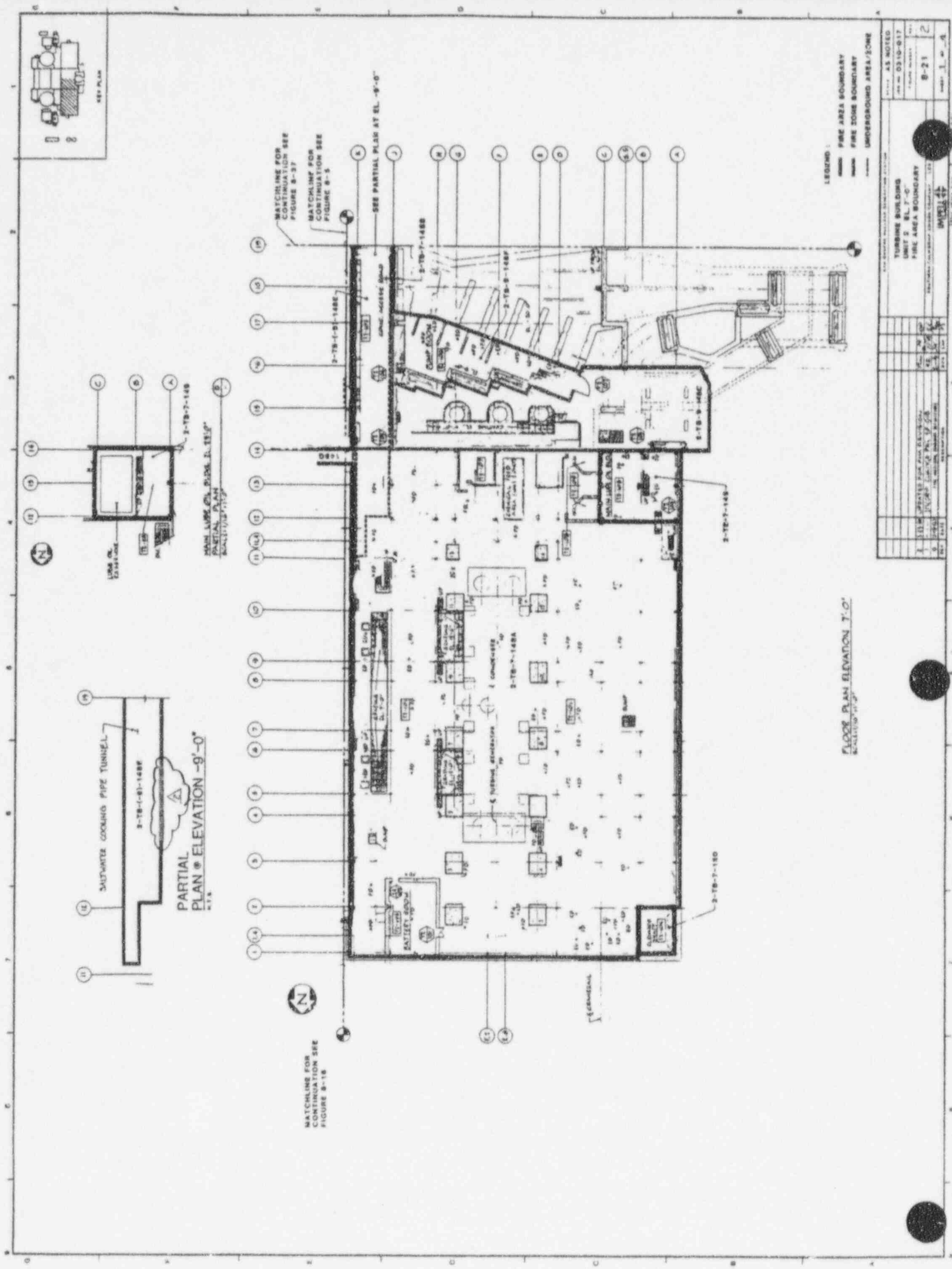

UNIT 2
SP CONTAINMENT BUILDING
MATCHLINE FOR CONTINUATION SEE FIGURE E-5
MATCHLINE FOR CONTINUATION SEE FIGURE B-1
2-TK-(-2)-161B
SURGE TANK ROOM
HEAT EXCH. ROOM
HEAT EXCH. ROOM
PIPING R.M.
HEAT EXCH. ROOM
HEAT EXCH. ROOM
PIPING TUNNEL
2-SE-(-2)-174
2-CT-(-2)-142B
2-SE-(-15)-138
2-SE-(-15)-139
2-SE-8-140B
CHEM. STOR
2-SE-8-140A
SURGE TANK RM
2-SE-8-141
PIPING R.M.
2-SE-(-5)-136A
2-TB-8-145G
2-TK-(-2)-161B
A/C ROOM
2-SE-(-15)-136
PIPING R.M.
LOWER ELECTRICAL TUNNEL
2-CT-(-2)-142B
2-TB-7-149A
2-TB-7-149A
MATCHLINE FOR CONTINUATION SEE FIGURE 8-21
MATCHLINE FOR CONTINUATION SEE FIGURE 8-25
PLAN AT ELEVATION 8'-0"
SCALE: 1/8"=1'-0"
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
KEY PLAN
SAFETY EQUIPMENT BLDG
UNIT 2 EL. 8'-0"
FIRE AREA BOUNDARY
AS NOTED
0310-017
8-16
SHEET 1 OF 4
IMPELL

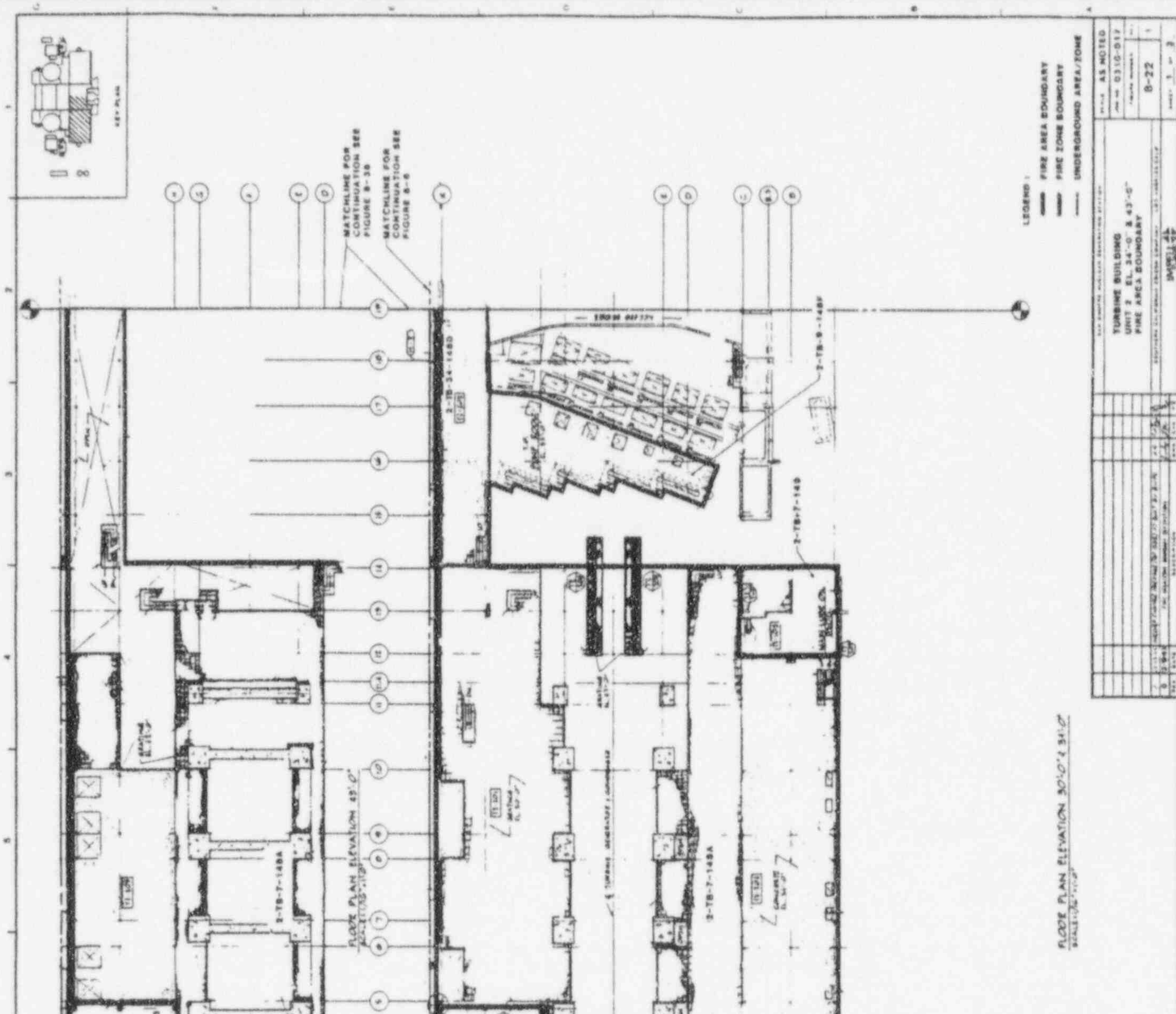

PLAN AT ELEVATION 30'-6"
SCALE: N.T.S.
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
SAFETY EQUIPMENT BLDG
UNIT 2 EL. 30'-6"
FIRE AREA BOUNDARY
8-17
KEY PLAN
UNIT 2
CONTAINMENT BUILDING
ELECT TUNNEL
WATER CONTROL ROOM
STAIR S3
ELEVATOR SHAFT
EQUIPMENT ACCESS ROOM
MATCHLINE FOR CONTINUATION SEE FIGURE 8-23


UNIT 2
PLAN AT ELEVATION 50'-6"
PARTIAL PLAN @ ELEVATION 70' - 0"
PARTIAL PLAN @ ELEV. 70'-0"
ELEVATOR SHAFT
STAIR A1
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
8-18

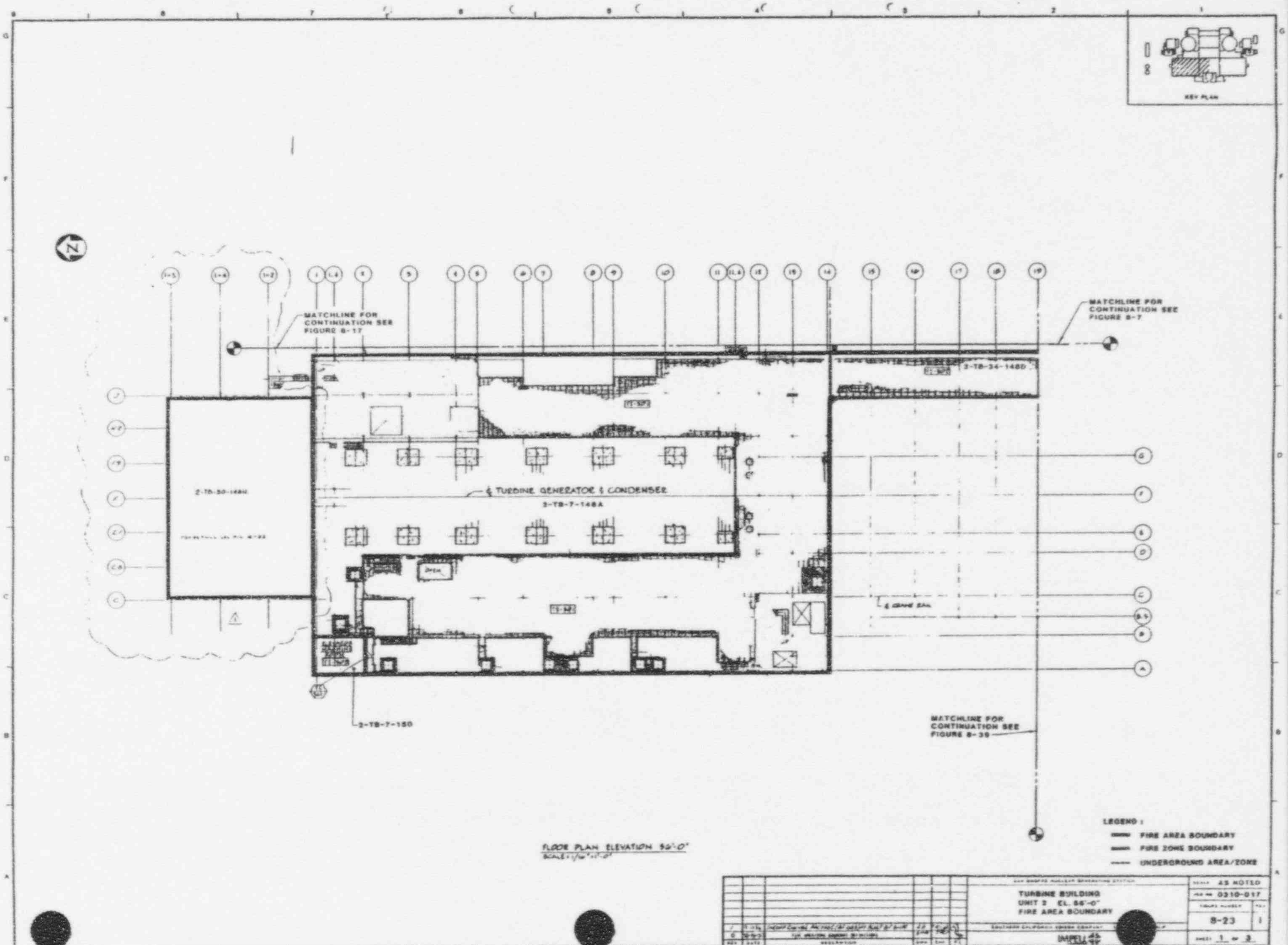

KEY PLAN
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
FUEL HANDLING BLDG
FIRE AREA BOUNDARY
8-19
MATCHLINE FOR CONTINUATION SEE FIGURE 8-10
MATCHLINE FOR CONTINUATION SEE FIGURE 8-1
MATCHLINE FOR CONTINUATION SEE FIGURE 8-12
MATCHLINE FOR CONTINUATION SEE FIGURE 8-2
VOID SPACE
SPENT FUEL POOL
2-FH-17-123
2-FH-17-122
2-FH-15-124
2-FH-15-125
2-CT-16-142C
2-FH-30-127
2-FH-30-128
2-FH-30-129
2-PE-30-2C
FLOOR PLAN EL 17'-6"
FLOOR PLAN EL 30'-0"


FLOOR PLAN EL. 45'-0"
FLOOR PLAN EL. 63'-6"
SECTION STAIR NO. 1 UNIT 2
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
KEY PLAN

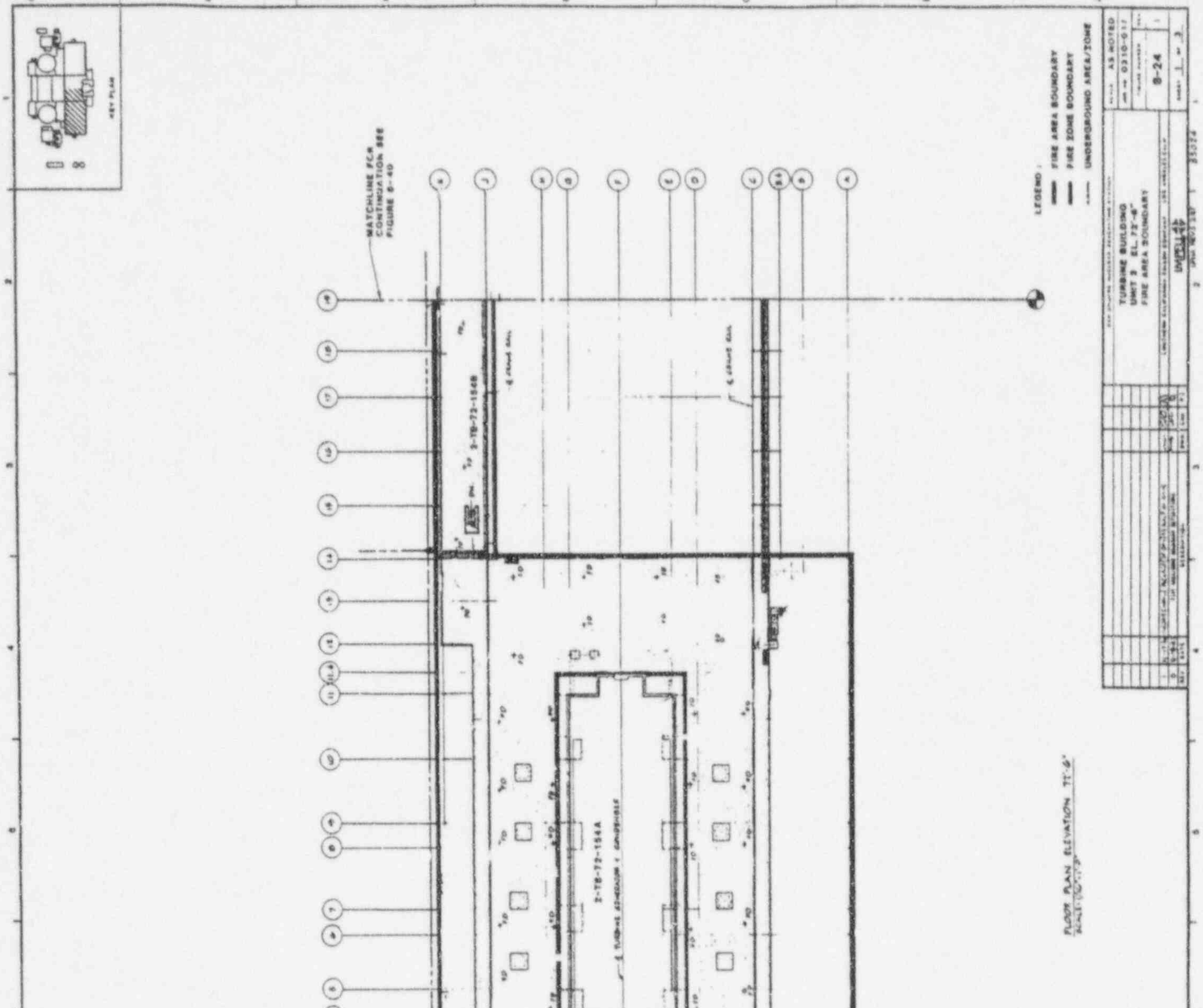

PARTIAL PLAN @ ELEVATION -9'-0"
SALTWATER COOLING PIPE TUNNEL
FLOOR PLAN ELEVATION 7'-0"
MATCHLINE FOR CONTINUATION SEE FIGURE 8-18
SEE PARTIAL PLAN AT EL. -9'-0"
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
TURBINE BUILDING
FIRE AREA BOUNDARY
8-21

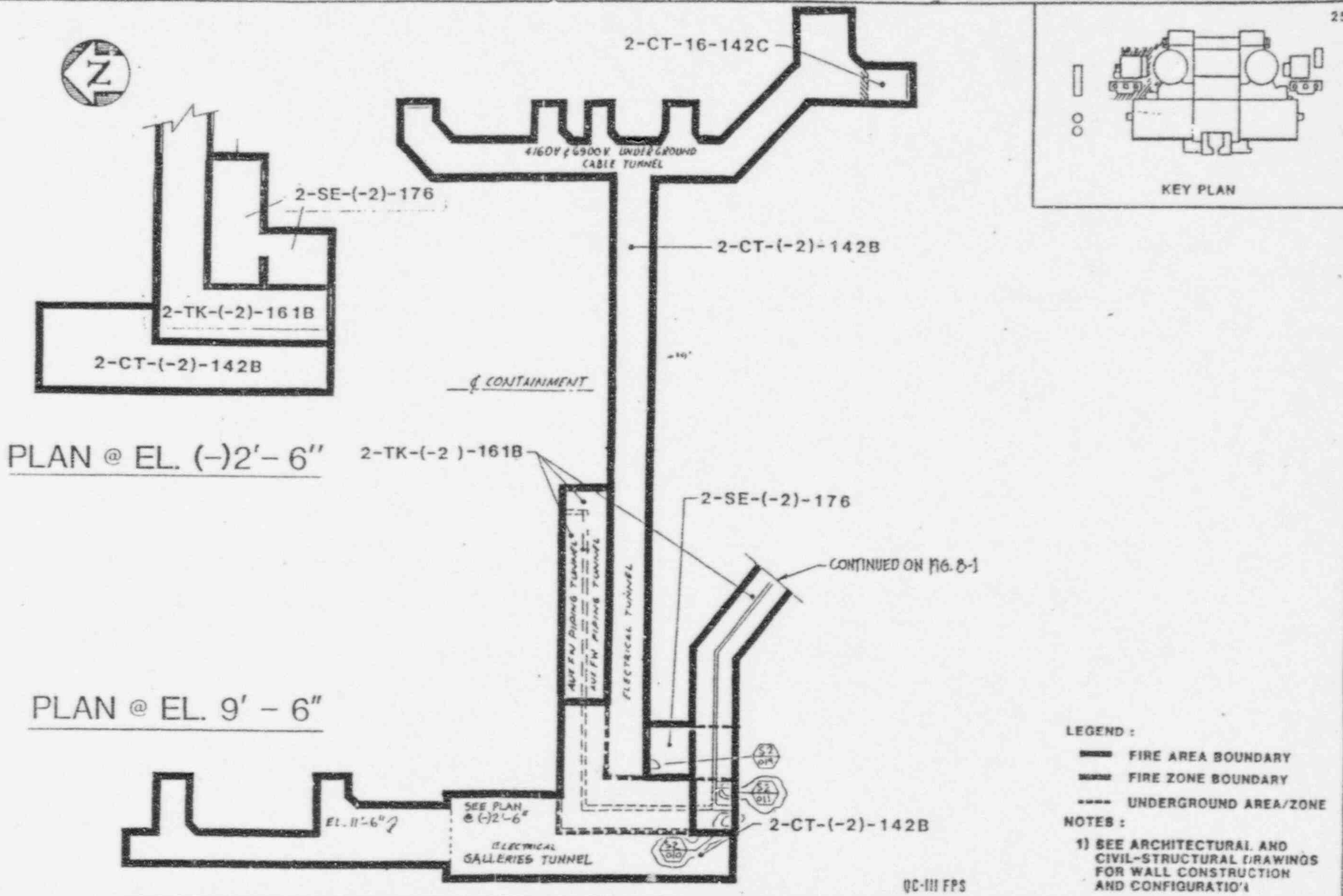


KEY PLAN
MATCHLINE FOR CONTINUATION SEE FIGURE 8-38
MATCHLINE FOR CONTINUATION SEE FIGURE 8-8
FLOOR PLAN ELEVATION 45'-0"
FLOOR PLAN ELEVATION 30'-0" & 34'-0"
LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
TURBINE BUILDING
UNIT 2 EL. 34'-0" & 43'-0"
FIRE AREA BOUNDARY
AS NOTED
8-22



KEY PLAN
MATCHLINE FOR CONTINUATION SEE FIGURE 8-17
MATCHLINE FOR CONTINUATION SEE FIGURE 8-7
MATCHLINE FOR CONTINUATION SEE FIGURE 8-39
2-TB-30-148H
2-TB-34-148D
℄ TURBINE GENERATOR & CONDENSER
2-TB-7-148A
2-TB-7-150
FLOOR PLAN ELEVATION 56'-0"
LEGEND :
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
TURBINE BUILDING
UNIT 2 EL. 56'-0"
FIRE AREA BOUNDARY
AS NOTED
0310-017
8-23
1
SHEET 1 OF 3


LEGEND:
FIRE AREA BOUNDARY
FIRE ZONE BOUNDARY
UNDERGROUND AREA/ZONE
TURBINE BUILDING
UNIT 2 EL. 72'-6"
FIRE AREA BOUNDARY
AS NOTED
8-24
MATCHLINE FOR CONTINUATION SEE FIGURE 8-40
MATCHLINE FOR CONTINUATION SEE FIGURE 8-8
2-TB-72-154B
2-TB-72-154A
FLOOR PLAN ELEVATION 72'-6"
KEY PLAN



REV.	DATE	DESCRIPTION	QA	D.	CHK.	P.E.
3	1-22-90	AS BUILT BY FLUOR; INC. CHG. PKG. SF-101		S.S.		
2	1-21-88	INCORPORATE CHANGE PACKAGES SF-082 AND SF-084				
1	5-5-88	UPDATED FOR FHA REVISION				
0	12-19-83	FIRE AREA/ZONE BOUNDARY DEFINITIONS				

SAN ONOFRE NUCLEAR GENERATING STATION

CABLE TUNNEL AREA
UNIT 2 EL.9'-6"
FIRE AREA BOUNDARY

SOUTHERN CALIFORNIA EDISON COMPANY LOS ANGELES, CALIF.

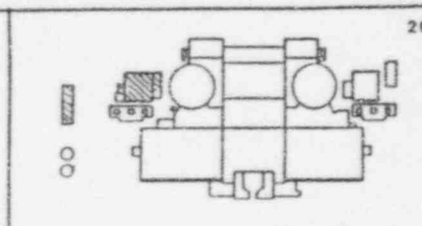
IMPELL

SCALE : N.T.S.

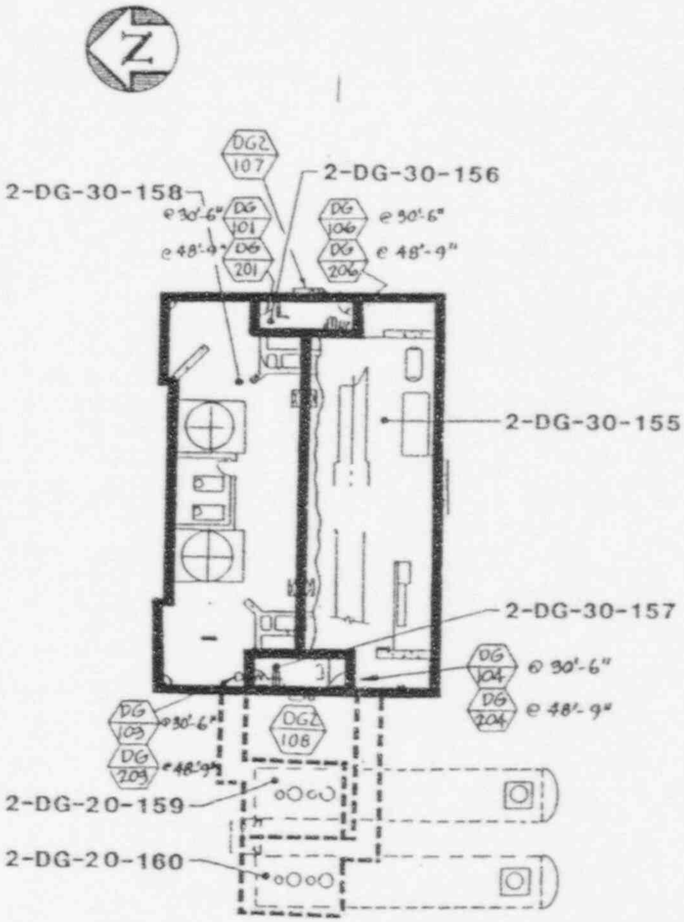
JOB NO. 0310-017

FIGURE NUMBER 8-25 REV. 3

1 of 3

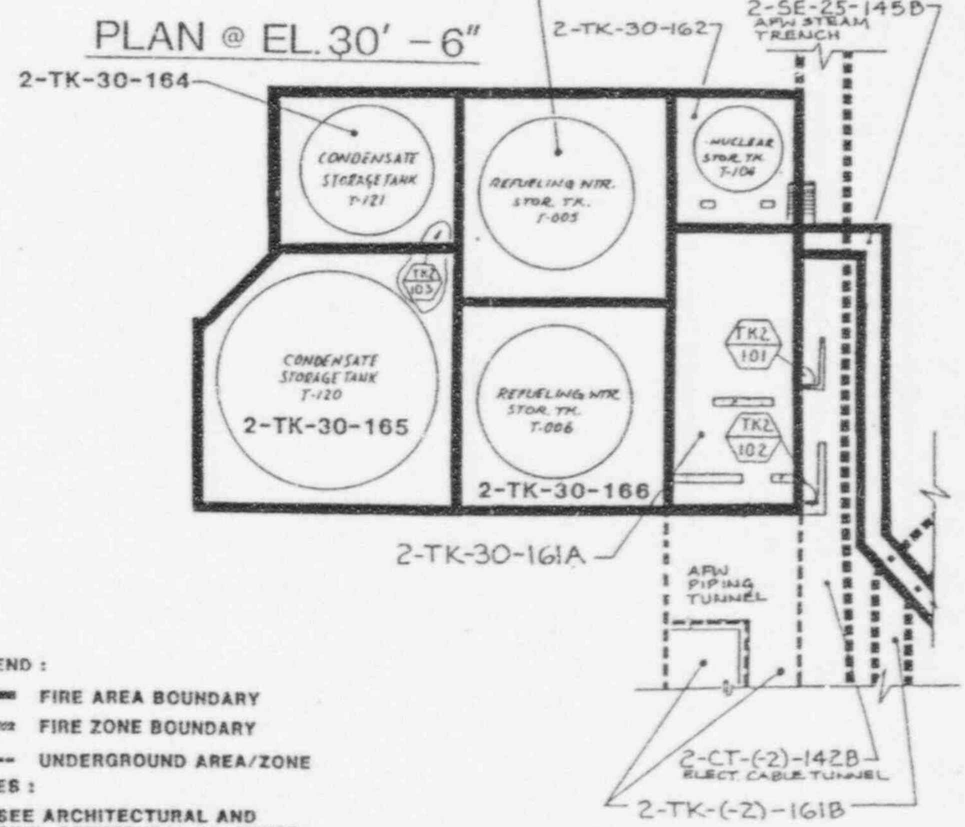


KEY PLAN



PLAN @ ELEVATION 30' - 6"

PLAN @ EL. 30' - 6"



LEGEND :

- FIRE AREA BOUNDARY
- FIRE ZONE BOUNDARY
- UNDERGROUND AREA/ZONE

NOTES :

1) SEE ARCHITECTURAL AND CIVIL-STRUCTURAL DRAWINGS FOR WALL CONSTRUCTION AND CONFIGURATION

QC-111FPS

REV.	DATE	DESCRIPTION		DWN.	CHK.	P.E.
3	1-22-90	AS BUILT BY FLUOR, INC. CHG. PKG SF-086 ADDED QC				
2	5-10-88	UPDATED FOR FHA REVISION				
1	11-20-85	INCORP. CHG. PKG. SF-007		RR		
0	12-19-83	FIRE AREA/ZONE BOUNDARY DEFINITIONS				

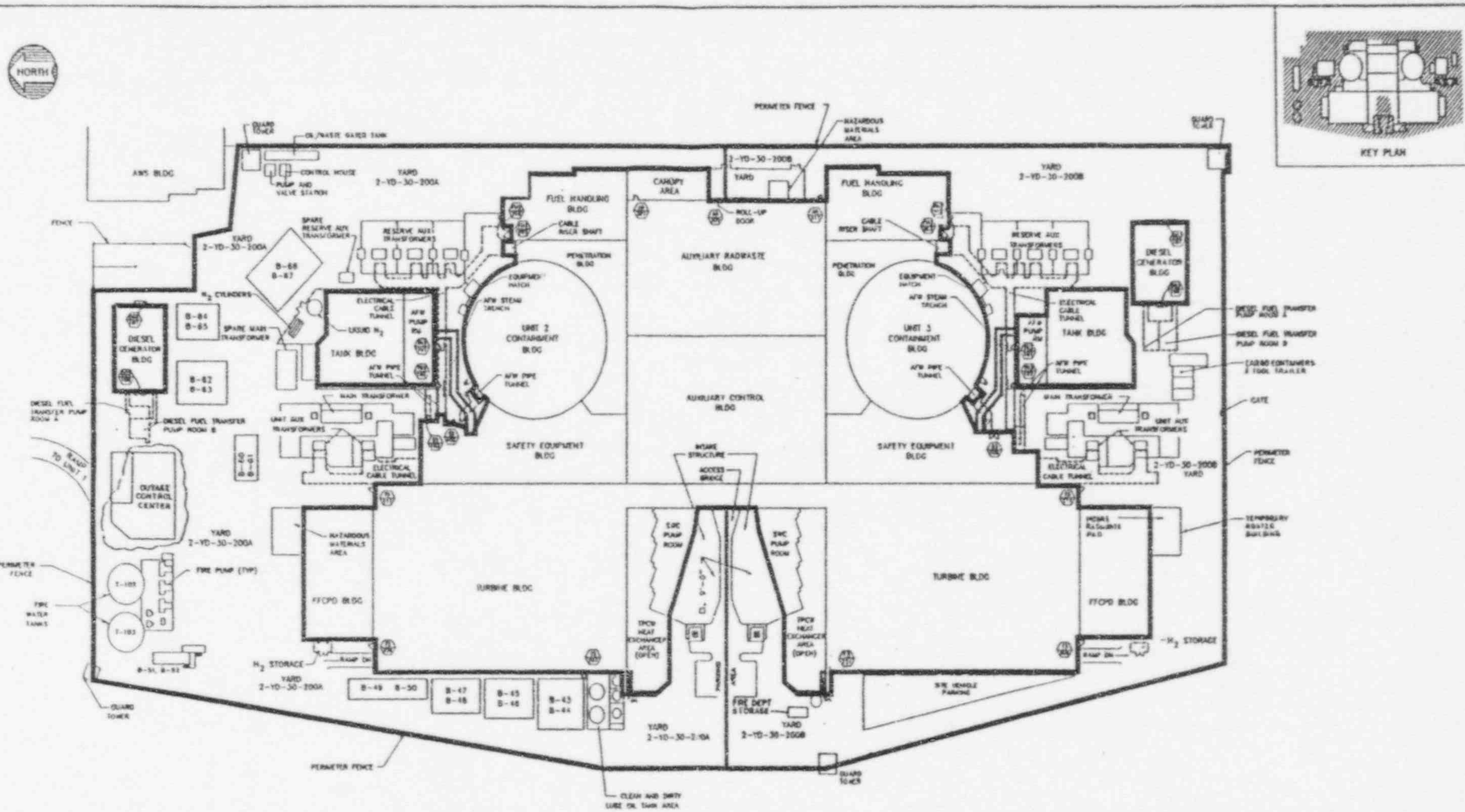
SAN ONOFRE NUCLEAR GENERATING STATION

DIESEL GEN. BLDG. & TANK BLDG.
UNIT 2 EL. 30'-6"
FIRE AREA BOUNDARY

SOUTHERN CALIFORNIA EDISON COMPANY LOS ANGELES, CALIF.

IMPELL

SCALE: N.T.S.
JOB NO. 0310-017
FIGURE NUMBER 8-26 REV. 3
SHEET 1 OF 4



PLAN @ ELEVATION 30'-0"

SCALE: N.T.S.

LEGEND:
- FIRE AREA BOUNDARY
- FIRE ZONE BOUNDARY
- UNDERGROUND AREA/ZONE

QC-IIIFPS

3	12-15-92	INCORP. DCN 3, AS BUILT.	LGS			
2	1-6-92					
1		FIRE AREA BOUNDARY				
REV	DATE	DESCRIPTION		DWN.	CHK.	P.E.

SAN ONOFRE NUCLEAR GENERATING STATION

YARD
UNITS 2 and 3 EL. 30'-0"
FIRE AREA BOUNDARY

SOUTHERN CALIFORNIA EDISON COMPANY LOS ANGELES, CALIF.

SCALE AS NOTED
JOB NO. 0310-189
FIGURE NUMBER 8-26A REV 3
SHEET 1 OF 3

UFHA REV. 9 2/93 83316

5. HIGH WINDS, FLOODS AND OTHER EXTERNAL EVENTS

Section 5 contains the evaluation of all external initiating events except earthquakes and internal fires. Events such as high winds, floods, accidents at nearby facilities, transportation accidents, external fires, volcanic activities and meteor strikes are included in this section. This section is organized into the following subsections:

5.1 BACKGROUND AND METHODOLOGY

External events posing significant risk had been considered in the original plant design of San Onofre Units 2 and 3, ensuring that the nuclear risk for the general public is acceptable. Section 5 of NUREG-1407 [5-1] requires a review of the plant-specific hazards and licensing bases with respect to external events, including the resolution of each event.

If significant changes in the bases of the plant license have occurred since issuance of the license, conformance of the current configuration to the Standard Review Plan must be verified. A detailed assessment is required for events with a core damage frequency larger than 1E-6 per year.

A review was performed to verify that the plant is still in conformance, and that no significant changes have occurred that could affect the original design conditions. Each of the external events were reviewed and compared to the FSAR analysis to verify continuing conformance. A confirmatory plant walkdown was performed concentrating on outdoor facilities that could be affected by external events such as high winds, onsite storage of hazardous materials, and offsite developments.

Many of the external events discussed in this section have a much lower frequency of occurrence at the San Onofre plant site than 1E-6 per year. For example, the likelihood of avalanches or volcanic activities affecting the operation of a nuclear power plant in Southern California is negligible. Other events may occur more frequently at San Onofre but their effect would not be severe enough to jeopardize safe operation of the plant. Some consequences of external events such as loss of offsite power caused by lightning had been included in the Individual Plant Examination and are, therefore, omitted from the IPEEE.

The procedural guidelines of NUREG-1407 specify that, besides earthquakes and fires, three types of external events must be assessed in detail for each plant site. These events are

- high winds and tornados,
- external floods, and
- transportation and nearby facility accidents.

The remainder of the external events can be screened by verifying the absence of site-specific external phenomena that pose a significant additional risk to the general public. Because of widely varying magnitudes of the external event frequencies and consequences, NUREG-1407 recommends progressive screening to eliminate events that do not contribute significantly to the core damage frequency.

5.2 PLANT AND SITE FEATURES RELEVANT TO EXTERNAL EVENTS

5.2.1 LOCATION AND NATURAL CHARACTERISTICS OF THE SITE

The San Onofre Nuclear Generating Station (SONGS) is located approximately 62 miles southeast of Los Angeles and 51 miles northwest of San Diego on the Southern California Coast. Its latitude is 33° 22' 10" and the longitude 117° 33' 30". The area is characterized by a gently sloping coastal plain that extends for about a half mile from the sea cliffs at the beach to the San Onofre Mountain. In the vicinity of the site, the cliffs reach a height of up to 100 feet. Portions of the coastal plain near the beach have been eroded by ephemeral streams, forming several deep barrancas.

Sparse coastal strand vegetation grows on the sandy beach at the base of the seacliffs. The coastal plain east of the plant site supports a variety of sage scrubs and grassland vegetation. No naturally growing trees can be found within one mile of the San Onofre plant. Areas that have been developed or disturbed are almost devoid of vegetation.

The site is about 4500 feet long and 800 feet wide, encompassing an area of 84 acres. Units 2 and 3 are built on approximately 53 acres of it. The reactor site had been cut out of the cliffs and has a grade elevation of 30 feet above mean lower low water (mllw). Together with the decommissioned Unit 1, Units 2 & 3 form a single site located near the north-west corner of the Camp Pendleton Marine Corps Reservation in San Diego County. The property upon which the power station is built has been leased from the United States Government. The nearest privately owned land is at a distance of about 2.5 miles from the site.

At the southwest boundary of the site, a permanent concrete seawall has been erected as protection against erosion by the Pacific Ocean. This wall is designed to withstand the design basis earthquake followed by a tsunami with coincident storm waves. The distance from the containment center to the exclusion area boundary varies from about 1,970 to 2,100 feet. There are no industrial, commercial, institutional or residential structures within the exclusion area boundary.

5.2.2 POPULATION

The population of the surrounding Camp Pendleton Marine Corps Base is expected to remain below 51,000 people. Administrative buildings and main housing areas are located 12 to 15 miles to the south-east of the San Onofre site. The distance to the closest living quarters at the base is about 1.5 miles. San Clemente is the nearest town to the San Onofre Nuclear Station at a distance of about 4 miles. In 1991, it had a population of about 42,000. Oceanside is 17 miles and San Diego 51 miles to the southeast of the plant. Section 2.1.3 of the FSAR contains more detailed data on the population distribution in the vicinity of the San Onofre site.

5.2.3 TRANSPORTATION ROUTES

Surface traffic routes and airways are located in the vicinity of the San Onofre Station. The distance to the closest shipping lane is more than 5 miles.

Highway 101 runs northwest to southeast directly adjacent to the plant on its east side, serving as the access road to SONGS and part of the San Onofre State Beach. The eight-lane Interstate Freeway 5 is located east of Highway 101 running parallel to it, and connects the metropolitan areas of Los Angeles and San Diego.

Between the two highways, the tracks of the Atchison, Topeka and Santa Fe Railroad run in the northwest/southeast direction. This line is currently used for cargo service and Amtrack passenger service.

Aircraft traffic in the vicinity of the plant is generally routed along four federal airways. The closest Airway is V-23 at a distance of one half mile from the plant, V-208-458 is about 7 miles and the two Vectors V-25-27 and J-1 are each at a distance of 12 miles from the plant. More information on aircraft traffic in the proximity of SONGS can be found in FSAR Section 2.2.2.5.

Commercial shipping lanes in the Pacific Ocean are more than 5 miles southwest of the plant.

5.2.4 STRUCTURES AND SYSTEMS SUSCEPTIBLE TO EXTERNAL EVENTS

Most buildings at SONGS are designed to protect equipment against the consequences of internal and external events. The design basis of the largest building, the containment, is the retention of fission products following a large LOCA. However, its massive prestressed concrete walls and leaktight design simultaneously protect systems in the building against external events.

Most safety-related systems at SONGS are protected by structures designed to withstand the design basis earthquake and tornados. Specifically, the auxiliary building, safety equipment building, fuel handling building, tank building, intake structure and diesel generator building protect all components housed within these structures against effects of external events. These structures are also designed to withstand blast overpressures of up to 7 psi.

Non-safety-related equipment is not required to be protected from external events. Loss of this equipment due to an external event can impact plant safety either by causing a trip (for example, loss of offsite power or loss of main condenser), or by unavailability of a mitigation system alternative (loss of main feedwater or a reserve auxiliary transformer). The systems belonging to this category at San Onofre are listed below:

- switchyard
- transformers (main, auxiliary and auxiliary reserve)
- power conversion system (main feedwater, condensate)
- fire water pumps and tanks
- normal ventilation and air conditioning systems
- normal compressed air system

Although seismic category I structures are designed strong enough to withstand the effects of all external events, it is possible that harmful substances or water could flow through openings into the structures, damaging vital equipment and/or making a building uninhabitable. These hazards will be discussed in the following sections.

5.3 SCREENING OF EXTERNAL EVENTS

The basis for the initial screening process of external events was a comprehensive collection of all pertinent data of natural and man-made external hazards. This included the updated San Onofre FSAR, extensive background information and more recent data where changes had occurred. A list of all external hazards from NUREG/CR-2300 [5-2] was adapted to reflect the conditions at the San Onofre Station. Data on the geologic, hydrologic and meteorological characteristics of the region as well as present and projected industrial activities and traffic densities in the vicinity of San Onofre were reviewed.

5.3.1 SCREENING METHOD

Five screening criteria were used to exclude insignificant risk contributors from more detailed evaluations. Four criteria have been adopted from the PRA Procedures Guide NUREG/CR-2300 and are listed below:

An event is considered to pose an insignificant risk if:

1. the event has the same or smaller damage potential than events for which the plant has been designed. This requires an evaluation of the plant design bases to estimate the resistance of plant structures and systems to the external event.

2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is small compared to those for other events. In this case, the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as not significantly influencing the total risk.

3. The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides and volcanic eruptions.

4. The event is included in the definition of another event. For example, the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility, or transportation accidents.

A fifth criterion has been added:

5. The event develops slowly and there is sufficient time to eliminate the source of the threat, or to take precautionary action to mitigate the consequences.

5.3.2 SCREENING RESULTS

These criteria were used in the screening process to reduce the number of external initiating events that have to be evaluated in detail for the IPEEE. Except for high winds, tornados, external floods and accidents caused by industrial, military and transportation facilities, all "other" external events have been eliminated as significant risk contributors by one or more of the screening criteria. The results are listed in Table 5.3-1.

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Aircraft Impact	--	See Section 5.6.3
Avalanche	3	The topography and meteorology of the site are such that avalanches are not possible.
Biological Events	5	UFSAR 10.4.5.2.3: The circulating water system is periodically heat treated to control marine growth. A portion of the circulating water is recirculated to raise water temperature while reversing flow in the intake and discharge conduits via a recirculation gate system. Sodium hypochlorite is also used to reduce marine growth in the intake structure. NaOCl is injected into the intake structure between the saltwater cooling pumps and the traveling screens. Traveling bars and traveling screens are located in the intake structure (upstream of SWC and circulating water pump suctions) to remove incoming marine life and debris. The screens are normally stationary until actuated automatically by debris buildup, or manually by the operator. Over the lifetime of the plant, severe winter storms have caused a buildup of uprooted kelp or marine grass on the intake screens on a few occasions. Several times, the debris accumulated at the screens caused enough flow resistance in the screens to lower the water level on the circulating water pump side by a foot or more. Plant operators were always able to remove the buildup during the event to prevent a plant trip. Even if the unit would have been tripped by clogging intake screens, the impact of this biological event on the core damage frequency would have been taken into account by the adjusted number for the plant-specific frequency of reactor trips. The saltwater cooling system is designed to handle much lower water levels than have occurred in the past and any anticipated severe level reductions in the future. To have a severe impact on SWC system functionality, the water level would need to drop several feet. Because this is extremely unlikely, biological events are screened.
Coastal Erosion	1,5	UFSAR 2.5.1.2.5.4: A detailed study was conducted to define the active erosional processes at San Onofre. The effects of past erosion and recommendations for erosion control regarding short- and long-term stability of facilities were outlined. Existing slopes at Unit 1 demonstrate the long-term stability of the exposed material. Also, engineered tsunami wall block coastal erosional forces.
Dam Failure	3	No dams are located near the site.
Drought	1,5	UFSAR 2.4.13.1: Fresh water requirements at San Onofre are met by water obtained from local water agencies and therefore no water will be derived from aquifers beneath or in the vicinity of the site for plant-related use.
Earthquake	--	See Section 3

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
External Fire (including forest, brush or grass fire)	3	There are no major wooded areas close enough to the site to pose a significant fire hazard. The plant is bordered by an eight-lane highway, an access road and large paved parking lots that significantly reduce any fire hazard. UFSAR 2.2.3.1.4, 2.2.2.1.4: The effects of a serious fire were estimated using data provided by the U.S. Forest service for coastal areas such as Camp Pendleton. It is estimated the worst case Santa Ana fire will consume 3000 to 4000 acre/hr with the fire front moving up to 200 ft/min and a 30 mph offshore wind. This fire is considered more limiting than from other postulated offsite sources. The fire is estimated to provide 10 ton/acre of fuel and emission rates of 200 lb. carbon monoxide/ton, 40 lb. hydrocarbon/ton, and 50 lb. particulate/ton. The maximum range of concentrations resulting from these conditions are found to be well below acceptable toxicity limits. Therefore, offsite fires are not considered to present a credible hazard to the plant.
External Flood	--	See Section 5.5
Fire Inside Plant	--	See Section 4
Fog/Smog	4	Per UFSAR 2.3.2.1.5, the occurrence of smog at San Onofre has no effect on plant operations. The only impact of fog on the core damage frequency is caused by traffic accidents on the highways that could lead to explosions, flammable clouds or toxic gas releases. The frequency of these accidents takes the meteorological conditions in the vicinity of San Onofre into account since the transportation accident data are site-specific for a 10-mile section of Interstate Highway 5 in the vicinity of the San Onofre Plant.
Frost	1	The combined conditions of low temperature and high humidity necessary for the formation of a sufficient frost layer to affect the operation of a power plant are never reached close to the Pacific Ocean at this latitude. The impact on the core damage frequency is negligible. UFSAR 2.3.2.1.2: Temperatures below 40°F are rare and periods of over 10 years may pass with no temperatures below freezing along the coast. The early morning average relative humidities in the summer are near 68% and in the winter near 58%. The average relative humidity ranges from about 60% during the day to about 75% at night. Occasionally, during Santa Ana conditions, the influx of the dry desert air can drop humidities in the area to less than 10%. Extreme maximum relative humidity is 100% during fog and/or precipitation (UFSAR Sections 2.3.2.1.3 and 2.3.1.1).

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Hail	1	Hail, in itself rarely causes sufficient damage to be reported. The damage potential of hail on exposed safety equipment at SONGS is small and inconsequential.
High Summer Temperature	1,5	The small diurnal and annual temperature changes of the Pacific Ocean ensure reliable operation of the plant and make a shutdown of the plant due to high heat sink temperature extremely unlikely. The impact of a high ocean temperature on the core damage frequency is negligible. The small range in temperature from day to night and from winter to summer produces a very equable regime along the southern California Pacific coast. High temperatures along the coast, although infrequent are associated with Santa Ana winds which occur in the autumn. The mean values at San Onofre are 2 to 3C cooler than the mean values recorded at San Diego and Los Angeles. Both Los Angeles and San Diego recorded their highest maximum temperatures in September 1963 (per UFSAR): LA=110°F and San Diego= 111°F. The maximum temperature at the site may occasionally reach 100°F or more with extreme Santa Ana wind conditions. Occasional Santa Ana winds cause higher maximum temperatures during fall and early winter (UFSAR Sect. 2.3.2.1.2).
High Tide	1	The high tide water level is within the plant design limits and does, therefore, not affect safe operation of the plant. The spring high tide has a 10% probability of exceedance of 7.0 feet above mean lower low water. Two high tides and two low tides of differing amplitudes occur each day on the average. Extreme high still water levels have been estimated at +9.3 feet mllw. This is calculated by summing the astronomical tide of +7.0 ft, the isostatic anomaly (+0.33 ft) and the maximum surge of +1.98 ft (UFSAR 2.4.5.2.1, 2.4.5.2.5 and DBD-SO23-TR-HZ, 2.2.3.2.5).
Hurricane & High Winds	3	See Section 5.4
Industrial Facility Accident	--	See Section 5.6
Ice Cover	3,5	The mild climate and general lack of freezing temperatures in the region around San Onofre make ice formation highly unlikely and which is, therefore, not considered a credible event (UFSAR Sect. 2.4.7). Ice jams at the plant intake structure do not exist in Southern California (UFSAR Sect. 2.4.2.1).

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Intense Precipitation	--	See section 5.5
Landslide	3	The engineered slopes on the plant site are designed so that landslides are impossible. Some of the steeper slopes are not engineered to the same standards, however, sliding of these slopes would not affect plant safety because of their distance to safety-related structures. Consequently, landslides are not credible accident initiators that could lead to core damage. In general, landsliding is a significant feature along the steep bluffs for some 5 miles south of the plant (UFSAR Section 2.5.1.2.5.2). The slides vary in age from a few years to over 125,000 years and appear to have been the result of wave action undercutting seaward-dipping bedding planes of the Monterey Formation. The San Mateo Formation on which the plant site is situated has been shown to be one of the most stable formations and least susceptible to landslides. This conclusion is based, in part, on field mapping which shows no landslides at or near the site in the San Mateo Formation.
Lightning	1	Based on low frequency and low plant damage potential, lightning as initiating event that leads to a nuclear accident and core damage is not a credible scenario. Estimates have been made of the frequency of occurrence of lightning ground strikes in the vicinity of SONGS utilizing an equation to estimate the lightning flashes going to the ground from the frequency of thunderstorm days. The calculated monthly flash density has been found to be less than 0.005 flashes per square kilometer. Per UFSAR Section 2.3.1.2.3, the normal annual occurrence of thunderstorms in Los Angeles and San Diego is 3 days per year, resulting in a low frequency of lightning strikes that hit buildings at the San Onofre site. The damage potential of lightning at San Onofre is also very low since the buildings at the site are designed with protection against lightning.

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Low Water Level	1	Tsunami, surge-caused and seiche-caused maximum drawdown conditions will not affect the ability of safety-related features at SONGS to function adequately Winds that blow offshore at San Onofre would cause the greatest lowering of water as a result of surge. Surge drawdown would be the most pronounced during Santa Ana wind conditions. A maximum credible Santa Ana condition for San Onofre would produce northeast winds of 35 knots sustained for 12 hours. The greatest correspondent drawdown from the antecedent water level associated with maximum Santa Ana wind condition is -0.55 foot. (UFSAR Section 2.4.11.2.1) The most severe low water that could hypothetically be assumed would involve the worst tsunami drawdown combined with the hypothetical extreme low still water level. The extreme low still water level at SONGS is estimated to be -2.63 feet mean low level water (mllw). This is derived from a situation consisting of a severe Santa Ana wind condition, as discussed above, causing a 0.55 sea level depression, promptly following passage of a deep low-pressure center in a winter storm causing an isostatic sea level anomaly of -0.33 foot (UFSAR 2.4.5.2.2) and occurring simultaneously with the lowest probable astronomical tide of -1.75 feet mllw (UFSAR 2.4.5.2.1). The maximum high water level of +15.6 feet would also cause the worst tsunami drawdown, which would be -11.9 feet mllw at the coast. This incident could persist for only a few minutes and only under the improbable condition that all of the contributing influences occur and reach their limits simultaneously. The worst low water case described above would not affect the ability of safety-related features to function at SONGS. The circulating water system receives its cooling water from intakes located 3,330 ft from the protective seawall and at maximum inlet depth of -20.75 feet mllw. Worst tsunami draw-down at the offshore intakes is -4.0 feet (UFSAR Section 2.4.11.2.2). As noted below, the effect of 'seiche' has been extensively measured near SONGS and has been found to affect sea surface elevation by only 0.7 centimeter. It is therefore concluded that tsunami, surge-caused and seiche-caused maximum drawdown conditions will not affect the ability of safety-related features at SONGS to function adequately (UFSAR Sections 2.4.11.1 and 2.4.11.2.1)
Low Winter Temperature	1,3	Minimum temperatures below 40°F are rare and time spans of over 10 years may pass without the temperatures ever falling below the freezing point at the Pacific Coast near San Onofre.
Meteorite	2	The frequency of meteorites causing damage on the earth's surface is extremely low (less than 10^{-9}) per NUREG 1407/NUREG CR-5042, Supplement 2.
Pipeline Accident	--	See Section 5.6

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Release of Chemicals	--	See Section 5.6
River Diversion	3	Upstream diversions associated with rivers, where low flow has an impact on dependable cooling water sources, is not a factor at SONGS. Cooling water is exclusively supplied by the Pacific Ocean and through conduits which have been designed to supply the minimum of 4% total intake conduit flow required for emergency cooling during any conceivable accident (UFSAR 2.4.9).
Sandstorm/ Dust storm	3	The risk posed by potential sandstorms at San Onofre is very low and the frequency of core damage due to sandstorms negligible. The greatest potential for occurrence of blowing sand or dust exists during strong winds from the northwest due to the existence of eolian sand in the San Onofre Creek area. However, the presence of natural vegetation in this area greatly reduces the potential for the production of blowing dust. The occurrence of blowing dust or drifting sand from the beach is minimized since sand becomes wet from spindrift under strong western/southwestern wind conditions (UFSAR Section 2.3.1.2.10).
Seiche	1,4	Per UFSAR 2.4.5.2.4, some of the most detailed measurements and analyses of long-period waves (normal shelf searching background levels) over the continental borderland have been conducted near Oceanside, CA, about 17 miles southeast of SONGS. Seiche has been determined to affect sea surface elevations by only 0.7 cm, which is considered negligible for water level calculations for southern California. (See also UFSAR 2.4.11.2.1) Event is bounded by high tide and low water level events.
Snow	1	UFSAR 2.3.1.1: Measurable snow has never been recorded at a coastal location in southern California.
Storm Surge	1	The probable maximum winds associated with the maximum surge (and seiche) water levels at SONGS are caused by northeast Pacific tropical cyclones that reach the southern CA coast. The maximum surge water level hypothetically possible at SONGS was determined utilizing a hypothetical storm with a speed of 10 knots and wind speeds in excess of 110 knots. The maximum likely storm surge height has been determined to be +1.98 feet above the antecedent level. (This figure is the sum of the barometric surge of +1.20 feet and the maximum surge components derived from the simplified steady-state surge model of +0.78 foot.) Hence, it is concluded that large surges will not develop in the vicinity of SONGS. (UFSAR 2.4.5.2.3.3) This precludes the necessity for a detailed two-dimensional treatment of surge such as a surge hydrograph. Refer to "High Tides" above for the calculation of the extreme high still water level using the +1.98 feet surge.

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Tornado	--	See Section 5.4
Toxic Gas	--	See Section 5.6
Transportation Accidents	--	See Section 5.6
Tsunami	1	The probable maximum tsunami at the site is based on the hypothetical occurrence of an earthquake with a 7 foot vertical displacement of the sea floor located 5 miles offshore from San Onofre as the generating mechanism. The wave induced by this earthquake occurring during simultaneous high tide and storm surge would have a maximum run-up to elevation +15.6 feet mean lower low water (mllw) at the Unit 2 and 3 seawall. When storm waves are superimposed, the resulting run-up is to elevation +27.5 feet mllw. Tsunami protection for the Unit 2/3 site is provided by a reinforced concrete seawall constructed to elevation +30 mllw. Therefore, the tsunami has no adverse impact on SONGS. (UFSAR 2.4.5.3, 2.4.6.1, 2.4.6.6).
Turbine-Generated Missiles	2	An evaluation was performed to determine the probability of unacceptable consequences following postulated occurrence of a turbine generator missile (DBD-SO23-TR-HZ, Rev.1). Analysis indicates that the high pressure turbine missiles and generator missiles would be retained by their respective casings. Additionally, the probability of missile damage from each turbine-generator is less than 1E-4 per turbine per year (per Reg Guide 1.115). The acceptably low probability of unacceptable damage resulted in the existing design being acceptable without a turbine missile shield being required. (UFSAR 3.5.1.3)
Volcanic Activity	3	There are no known volcanos that would impact San Onofre.

TABLE 5.3-1
SCREENING OF EXTERNAL EVENTS FOR SONGS 2/3

Event	Applicable Screening Criterion	Remarks
Waves	1	Based on analysis of historical data, wave action will not generate water levels above the elevation at the top of the seawall (+30.00 feet mllw). UFSAR 2.4.5.3: Severe deep water storm waves determine the lowest and highest instantaneous water elevations in conjunction with long period phenomena (i.e., tide and storm surge). Approximately 60 storms that occurred between 1900 and 1967 were examined to determine the maximum wave action. Twenty-five of the most severe storms were selected. The deep water wave data were corrected for refraction and shoaling at SONGS, and also for island sheltering. An extrapolation of the data gives the 100-year highest individual shallow water wave of 46 feet. The highest hindcast wave was produced by the tropical storm of September 24-25, 1939, which was the only tropical storm in the last 75 years that followed such a trajectory as to produce severe waves in southern CA waters. The greatest shallow water wave height offshore at San Onofre during the 1939 storm was 43.8 feet. A hypothetical tropical storm was considered based on the concurrence of individual worst parameters. Having an optimum final trajectory from the south, the credible conditions for worst storm wave generation includes an effective fetch of 400 miles with wind speed of 50 knots for 24 hours. This would generate a significant deep water wave height of 34 feet. The highest individual wave, corrected for sheltering, shoaling, etc., at the site is calculated at 54 feet (shallow water height). Its associated wave period would be 13 seconds. It is estimated to be a 200-year return internal wave. The worst storm-generated wave of 54 feet would begin feeling the bottom at a distance offshore of approximately 11,000 feet and would completely dissipate by the time it reaches San Onofre beach. Inasmuch as the significant wave height is the average of the highest one-third of the waves present, then two-thirds of the waves would be lower, and on an annual basis, the 1% height exceedance for all waves would be about 6 feet. Thus, the calculated highest run up at the seawall of +27.5 feet mllw due to storm waves occurring during an extreme high water level of 15.6 feet mllw (including tsunami) is far more severe than the 1% exceedance surf height.

5.3.3 WALKDOWN

A special plant walkdown [5-3] was performed and documented by NSG staff who had previously reviewed the requirements of NUREG-1407 for other external events, reviewed information on the plant design hazards and licensing bases, and reviewed the resolution of each event. This walkdown was performed to verify that the plant conditions and configuration with respect to external hazards had not changed since the updated FSAR evaluation. The walkdown concentrated on outdoor facilities that could be affected by high winds, floods, external fires, onsite storage of hazardous materials, and offsite developments.

A checklist was prepared for the walkdown which listed those potential external events and associated mitigation features that were appropriate for visual review and assessment. Each of the external events were evaluated to determine whether the hazard or plant design feature was amenable to visual verification, and what aspects should be examined and confirmed. This review included items such as external flooding sources and targets, roof ponding potential, tornado missile sources and shielding, biological fouling, non-engineered slope failure potential, protection from external fires, and sources of toxic or flammable gases.

In summary, the confirmatory walkdown for other external events verified that there were no identifiable hazards that had not been considered in the FSAR and design basis analyses, and that mitigating design features were in place as documented in the analyses.

5.4 HIGH WINDS AND TORNADOS

All SONGS 2/3 seismic category 1 structures are protected against pressure loadings and missiles generated by tornados. The structures located above the ground that are not seismic category 1 were designed for wind loadings only.

5.4.1 WIND LOADINGS

The structures at SONGS 2 and 3 are designed to withstand forces resulting from wind velocities of up to 100 miles per hour (mph). For comparison, the fastest wind velocity logged at San Diego during 31 years was 51 mph and the highest peak gust recorded at Los Angeles in 25 years was 62 mph. The 100-year return period estimate for the fastest windspeed is 57 mph at Los Angeles, and 47 mph at San Diego. Both the Los Angeles International Airport as well as the North Island Air Station near San Diego (where the data were recorded) are located on flat land and, therefore, well exposed to wind. In contrast, the San Onofre site is excavated from the cliffs. Only the containments exceed the height of the surrounding terrain. Additional protection is provided by the San Onofre mountains in the East.

Measurements taken during the last three decades at the San Onofre site prove that safety margins used in the analysis of wind loadings are actually larger than those of the original FSAR calculations. The fastest wind velocity measured by the SONGS meteorological tower since 1971 at the 10 meter (33 ft) level was 47.4 mph. At a level of 40 meter (131 ft), the fastest wind velocity during the same period was 49 mph. Consequently, a 100-year return period maximum wind speed of 63 mph can be considered a conservative estimate for the San Onofre site. This conservative value is significantly lower than the design wind speed of 100 mph used for the structures at San Onofre.

Hurricanes with wind velocities of more than 72 mph have never been observed in the southwestern United States. Analyses of hurricane hazards for the San Onofre Station are, therefore, not required.

5.4.2 TORNADO LOADINGS

Tornado hazards in the mountainous regions of California are very low. Between 1950 and 1975, a total of 177 tornados were recorded in California. About 70% of them did little or no damage. Only one tornado with wind speeds in excess of 157 mph was observed in California. The majority of the tornados recorded in Southern California belonged to Class F1 with a maximum wind velocity of 73 to 112 mph. No F4 or F5 tornado (with velocities larger than 206 mph) occurred during a 25-year observation period.

An analysis of the tornado hazard at San Onofre Unit 1 completed in 1990 estimates the probability of a tornado strike to be about 4.6E-5 per year [5-4]. However, the only tornado capable of damaging safety-related equipment and buildings at SONGS 2&3 are F5 tornados. The probability of an F5 tornado strike is 8E-8/year. Therefore, the tornado design basis using a maximum total wind velocity of 260 mph (peripheral tangential wind speed of 220 mph plus 40 mph translational speed) is conservative.

The structural integrity of the seismic category 1 buildings was analyzed for three types of loads to determine the maximum stress in building walls. These loads were tornado-induced differential pressure, dynamic wind pressure and missile impacts from tornados. Equipment that is required to be available for safe shutdown has been protected from tornado impact by designing it to withstand wind forces and missiles from tornados or housing it within a structure designed to withstand tornados.

The pressure in the center of the tornado is modeled by a 1.5 psi pressure drop within 4.5 seconds, followed by a constant pressure for 3 seconds and a 1.5 psi pressure rise over 4.5 seconds. All buildings that are seismic category 1 can withstand an internal overpressure of 1.5 psi. Except for the containment which is designed for a much higher pressure difference, the overpressure in buildings would be reduced significantly by leakage through openings and vents during the pressure transient. Tornados are not considered a credible threat to the integrity of the plant. The switchyard and transformers are the most likely equipment to be damaged by tornado, resulting in a loss of offsite power. Since plant-specific data for this initiating event include external events, this contribution to the core damage frequency has been taken into account in the IPE.

5.5 EXTERNAL FLOODS

Two areas in the vicinity of the site were considered potential flood sources:

- The San Onofre Creek Basin
- The foothill drainage area east of the plant

The probable maximum flood (PMF) was selected as the design basis event. Based on the PMF analysis, one can conclude that the San Onofre Creek Basin does not pose a flooding hazard for the site. The distance from the mouth of the creek to the power plant and the topographical features between the two locations prevent the San Onofre Creek from flooding the site.

Prior to plant operation, analysis of the foothill drainage area indicated that it had a potential to flood the site during a design basis PMF. To preclude flooding of the site by drainage from the foothill area in the east, a berm was erected that would divert the water towards the San Onofre Creek. The berm is monitored for weather-induced erosion and other deterioration to maintain it in the as-designed condition.

Intense precipitation at the site itself has the potential to cause flooding of the San Onofre Station. During the winter months, major storms move through Southern California coastal areas. The probable maximum precipitation (PMP) for frontal systems at the San Onofre site was calculated using data from the U.S. Weather Bureau. However, the thunderstorm PMP causes higher flood levels than the frontal system PMP and was, therefore, selected as the design basis event.

The drainage system was assumed to malfunction (with all roof drains, floor drains and catch basins plugged) to determine the highest water level during the thunderstorm PMP. In this case, the upper site area will drain directly into the barranca and to the beach without affecting the rest of the site. Rain water collecting on the asphalt surrounding the power block is conveyed by the graded asphalt surface to the seawall from where it flows into the ocean. This includes the water from the building roofs flowing through scuppers in the parapets because of the plugged drains. No safety-related equipment of any building will be affected if a thunderstorm PMP occurs since water-tight barriers prevent water from reaching most safety-related areas. Negligible water, if any, would reach areas protected by leak limiting doors (such as the emergency chillers and saltwater cooling pumps).

Section 6.2.2.3 of NUREG-1407 refers to Generic Letter 89-22 which recommends an evaluation of the new PMP data and their effect on flooding and roof ponding. An evaluation of "Probable Maximum Precipitation Estimates, Colorado River and Great Basin Drainages" of the National Oceanic and Atmospheric Administration was completed in March 1990 for the San Onofre site. The conclusion was that the original data used in the FSAR were more conservative than the new PMP values and envelop the newer data. The FSAR analysis contains, therefore, a larger safety margin than required per NUREG-1407.

During the plant inspections and walkdown [5-3] performed for the IPEEE, some parapet scuppers designed for rainwater drainage from the auxiliary building roof were found to be permanently closed. Based on maximum probable precipitation data for thunderstorms (FSAR Section 2.4.2.3, Table 2.4-2), the highest water level on the roof caused by the flow restriction in the reduced number of parapet scuppers was calculated. Using conservative assumptions such as complete plugging of all regular drains on the roof, this level was determined to be less than 7 inches. The weight of 7 inches of rain water accumulating on the roof is about 36 lb/ft^2. This is significantly less than the roof design load of 250 lb/ft^2, indicating a large safety margin in the design of the Auxiliary Building roof.

5.6 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

Three major transportation routes and three pipelines pass within one thousand feet from the San Onofre Station. Shipping lanes for commercial vessels are at a minimum distance of 5 miles in the southwest direction from the plant. Four airways pass by the plant within a radius of 12 miles.

The only military or industrial facilities within five miles from the San Onofre Station are firing ranges of the U.S. Marine Corps at the Camp Pendleton Base and a quarry.

5.6.1 TRANSPORTATION ACCIDENTS

Section 2.2.3.1 of the SONGS 2/3 UFSAR contains a probabilistic analysis of accidents caused by the transportation of hazardous materials on Interstate Highway 5 and the railway between the highway and the site. Due to changing transport frequencies, shipment sizes, cargoes and accident probabilities, the NRC requires a tri-annual update of the analysis per Technical Specification 6.9.1.14. The last updated analysis was issued in December 1993 as "San Onofre Nuclear Generating Station 1993 Off-site Hazard Update" and covers road and rail traffic passing in the vicinity of San Onofre [5-5]. Potential releases of toxic chemicals jeopardizing the habitability of the control room, and releases of explosive or flammable chemical mixtures that are capable of damaging buildings and equipment have been evaluated.

5.6.1.1 Interstate Highway 5

The 1993 Off-Site Hazard Analysis Update was based on information collected by a roadside survey of the truck traffic on Interstate Highway 5 from September 7 to 20, 1993. Survey personnel were stationed at the south-bound and north-bound traffic weigh stations San Onofre in order to determine the contents of the trucks. About 1800 materials were identified. Based on the overall hazardous material shipment frequency during the two week period, an annualized shipment frequency was calculated. The updated shipment frequencies were divided by the original frequencies given in the FSAR. This ratio was multiplied by the original hazard frequency, resulting in an updated hazard frequency. The individual hazard frequencies, with exception of those protected by the TGIS, are summed to obtain the plant risk for toxic hazards.

All shipments with a potential to generate more than 7 psid overpressure at a plant structure were evaluated together as a single type of event and compared to the Standard Review Plan acceptance criterion. In a similar way, shipments capable of producing a fire that affects the nearest safety-related plant structure were evaluated as one type of event and the results compared with the Standard Review Plan criteria.

5.6.1.2 Railroad Accidents on the Atcheson, Topeka & Santa Fe Line

Data of railroad shipping hazards were evaluated in 1993 [5-5]. LPG rail shipments on the AT&SF track had increased from 1499 shipments in 1990 to 2230 shipments in 1992. As a consequence, the potential for railroad accidents affecting San Onofre has increased. The frequency of LPG explosions with pressure waves of more than 7 psi resulting in damage to safety-related structures was estimated in the 1993 report to be about 1.5E-7/year.

5.6.1.3 Transportation Results

The results of the analysis confirms that SONGS 2 & 3 meet the Standard Review Plan criteria. Explosions generating an overpressure of at least 7 psi following a transportation (highway and railway) accident have a frequency of 2.7E-7/year. Flammable vapor clouds released by a transportation accident that may potentially jeopardize plant safety have a frequency of 6.2E-7/year. These frequencies of accidents damaging safety-related plant structures are well below the Standard Review Plan criterion of 1E-6/year that applies to conservative accident evaluations.

5.6.2 MILITARY EXPLOSIVES

Based on information provided by the U.S. Navy and Marine Corps, the largest military shipment of explosives past San Onofre from January 1991 through August 1993 was 27,014 pounds. Conservative assumptions were used to calculate the maximum overpressure at safety-related plant structures in the event that the largest shipment would explode. The results indicate that the explosion of the largest ordinance shipment would produce an overpressure of 4.7 psi at the plant. This is less than the design limit of 7.0 psi for safety-related buildings. Therefore, an explosion of military ordinance on Interstate Highway 5 does not jeopardize the safety of the San Onofre Station.

5.6.3 AIRCRAFT IMPACT ON PLANT EQUIPMENT

Four airways transit the airspace in the vicinity of San Onofre. The closest is Airway V-23 at a distance of 0.5 statute miles. Airway V-208-458 is located 7 miles from the plant and airways V-25-27 and J-1 are both 12 miles from San Onofre.

The closest air field in the vicinity of SONGS is a military air strip at Camp Pendleton located about 13 miles from the plant. The Orange County Airport 27 miles from San Onofre is the closest commercial airport. There are many general aviation airports in San Diego and Orange County. The nearest airports for general aviation are in the vicinity of Oceanside about 15 miles southeast and Fallbrook 17 miles east of SONGS.

In order to update the 1975 information given in Table 2.2-2 of the UFSAR, more recent data concerning aircraft operations were requested from the Aeronautics Division of the California Department of Transportation and the Southern California Office of the FAA. In response, these agencies provided new information on aircraft operations at airports within a 50-mile radius around San Onofre [5-6, 5-7].

The new data indicate that the air traffic volume generally increased. However, several airports have been or will be closed, reducing the overall increase. Military operations have been curtailed in comparison to the level in 1975. For example, Miramar Naval Air Station had 789,795 aircraft operations in 1975 and only 144,440 in 1993. For the year 2000, the estimated number of aircraft operations is 209,863 (see Table 5.6-1). All six military air stations together now have an operations volume that is slightly more than half of what it was in the Year 1975 and will be less than half in the Year 2000 (see Table 2.2-2 of the UFSAR).

Section 2.2.2.5.3. of the FSAR describes the aircraft activities in the vicinity of San Onofre for general aviation, commercial airliners and high-speed business jets, and military aircraft. Most of the information concerning the location and direction of the airways remains accurate. However, air traffic control in Southern California has been consolidated into a single TRACON facility (Traffic Control Center) located in San Diego. Before this change, there were different TRACON facilities in El Toro, San Diego and three other Southern California locations.

Because of a single, consistent data base and automatic flight data recording with digital computers, a more realistic estimate of the number of aircraft operations close to San Onofre has been developed. Based on an evaluation by the Southern California TRACON, San Diego, the number of aircraft operations near San Onofre had been overestimated significantly in the evaluation contained in the FSAR. The new estimate by the San Diego TRACON is approximately 92,300 aircraft operations per year in the vicinity of San Onofre. This number includes an estimated 25% of aircraft flying in the vectors near San Onofre that do not contact the Southern California TRACON and do not register on RADAR because of low altitude. Consequently, they are not counted by the automatic system.

The total number of aircraft operations given in the FSAR is 198,848 operations per year. This is more than twice the new, more realistic estimate. Using recent aircraft operations data, one can conclude that the risk posed by aircraft impact on the San Onofre facilities is small and does not exceed the original, more conservative estimate given in the FSAR.

5.7 CONCLUSION

An evaluation of the risk caused by other external events such as high winds, floods, and accidents at nearby industrial, military and transportation facilities has been conducted in accordance with NUREG-1407. It supports the generic assessment of NUREG/CR-5042 and Supplements 1 and 2 that the risk posed by these external events is not significant.

A screening process was used to eliminate events with an extremely low probability such as meteorites and volcanic eruptions from the analysis. Changes indentified in the original FSAR analysis required additional evaluation. New data or evaluations for tornados, precipitation and various transportation modes were available and applied to the analysis of winds, floods and ground or air transportation. A special walkdown was performed to verify that plant conditions as documented in the last updated FSAR had not changed with respect to external events.

The results of this external events analysis demonstrate that the plant meets the criteria of the 1975 Standard Review Plan, and that analyzed and screened events do not pose a significant risk. In accordance with NUREG-1407, this study demonstrates that "no other plant-unique external event is known that poses a significant severe accident threat within the context of the screening approach for 'High Winds, Floods and Nearby Facility Accidents'".

TABLE 5.6-1
AIRCRAFT OPERATIONS AT AIRPORTS WITHIN 50 MILES FROM SAN ONOFRE

AIRPORT TYPE Name	**Distance (Miles)**	**Sector**	**Operations in 1993 (Actual)**	**Year 2005 (Estimate)**
COMMERCIAL AIRPORTS				
Orange County (J.W.)	27	NW	494,477	730,926
Ontario	45	N	154,850	549,830
Long Beach	45	NW	4[illegible]0,774	526,438
San Diego (Lindbergh)	48	SSE	213,022	384,853
Los Angeles Internatl.	63	NW	682,161	1,106,173
MILITARY AIRSTATIONS				Year 2000
Camp Pendleton MCB	13	ESE	85,877	176,800
El Toro MCAS	20	NNW	111,000	Closed
Santa Ana MCAS	29	NW	109,400	Closed
March AFB	40	NNE	89,913	77,000
Los Alamitos NAS	41	NW	111,600	112,000
Miramar NAS	42	SE	144,440	209,863
GENERAL AVIATION AIRPORTS				Year 2005
Oceanside	15	SE	78,000	103,263
Fallbrook	17	E	25,000	43,584
Skylark	22	NE	6,000	11,645
Palomar	23	SE	224,479	309,652
Rancho California	25	ENE	Closed	Closed
Perris Valley	30	NNE	25,000	29,000
Corona	35	N	110,000	205,532
Meadow Lark	35	NW	Closed	Closed
Fullerton	40	NW	170,000	169,030
Hemet-Ryan	40	NE	110,670	129,408
Riverside	40	N	100,000	154,662
Chino	45	N	268,000	255,253
Fla-Bob	45	N	36,400	68,550
Ramona	45	ESE	121,252	97,814
Montgomery	47	SSE	249,936	294,724
Brackett	50	NNW	235,000	230,234
Cable	50	N	88,000	188,900
Gillespie	50	SE	191,660	202,510
Tri-City	50	NNE	Closed	Closed
Avalon	50	W	48,020	56,581

5.8 REFERENCES

5-1 "Procedural and Submittal Guidance for the IPEEE," NUREG-1407, U.S. Nuclear Regulatory Commission, June 1991.

5-2 "PRA Procedures Guide," NUREG/CR-2300, January 1983.

5-3 "Confirmatory Walkdown Report for IPEEE, Other External Events," San Onofre Nuclear Generating Station, SCE, Nuclear Safety Group, February 8, 1995

5-4 "Tornado Hazard Review for San Onofre Unit 1," Final Report, ERIN Engineering Report, ERIN Engineering and Research, June 1990.

5-5 "1993 Off-Site Hazards Update," San Onofre Nuclear Generating Station, NSG/PRA Report PRA-2/3-93-008B, December 1993.

5-6 "Projected Operations for Airports," Duane Ferguson, Division of Aeronautics, California Department of Transportation, December 15, 1994.

5-7 "Aircraft Operations at Military Airports," William Frank, Federal Aviation Administration, July 7, 1995.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE PROGRAM ORGANIZATION

The SONGS 2/3 IPEEE team is shown in Figure 6.1-1. The Nuclear Safety Group (which is part of the Safety Engineering Section of the Nuclear Oversight Division) and the Civil Engineering Group [which is part of the Nuclear Engineering Design Organization (NEDO)] cooperatively managed and conducted the IPEEE program at SCE. Each were responsible for program budget, schedule, and resource allocation, in addition to the responsibility for resolution of major technical issues. The lead analysts for each initiating event (i.e., seismic, fire, and high winds) managed the day-to-day development of the analysis and addressed issues on a continuous basis. Mr. David Moore of EQE International provided overall technical guidance as the project technical advisor.

The allocation of resources and effort was based on the initiating events: seismic events, internal fires, and high winds, floods, and others.

<u>Seismic</u>:

The Nuclear Safety Group (NSG) and the NEDO-Civil Group combined to complete the seismic analysis. The Nuclear Safety Group developed the seismic equipment and relay lists, participated in a majority of the walkdowns, performed relay chatter evaluation and screening analyses, completed a seismic human reliability assessment, and completed the Level I and II quantification. NUS Corporation provided technical assistance with the Level II development. The NEDO-Civil Group and EQE combined to perform the seismic capacity walkdowns and seismic fragility calculations for components on the seismic equipment and relay lists. EQE also performed the response analyses of structures and provided overall seismic technical guidance.

Seismic hazard characterization was a collaborative effort between Geomatrix Consultants (seismic source characterization), Woodward-Clyde Consultants (attenuation relationships), and Risk Engineering, Inc. (seismic hazard characterization and integration).

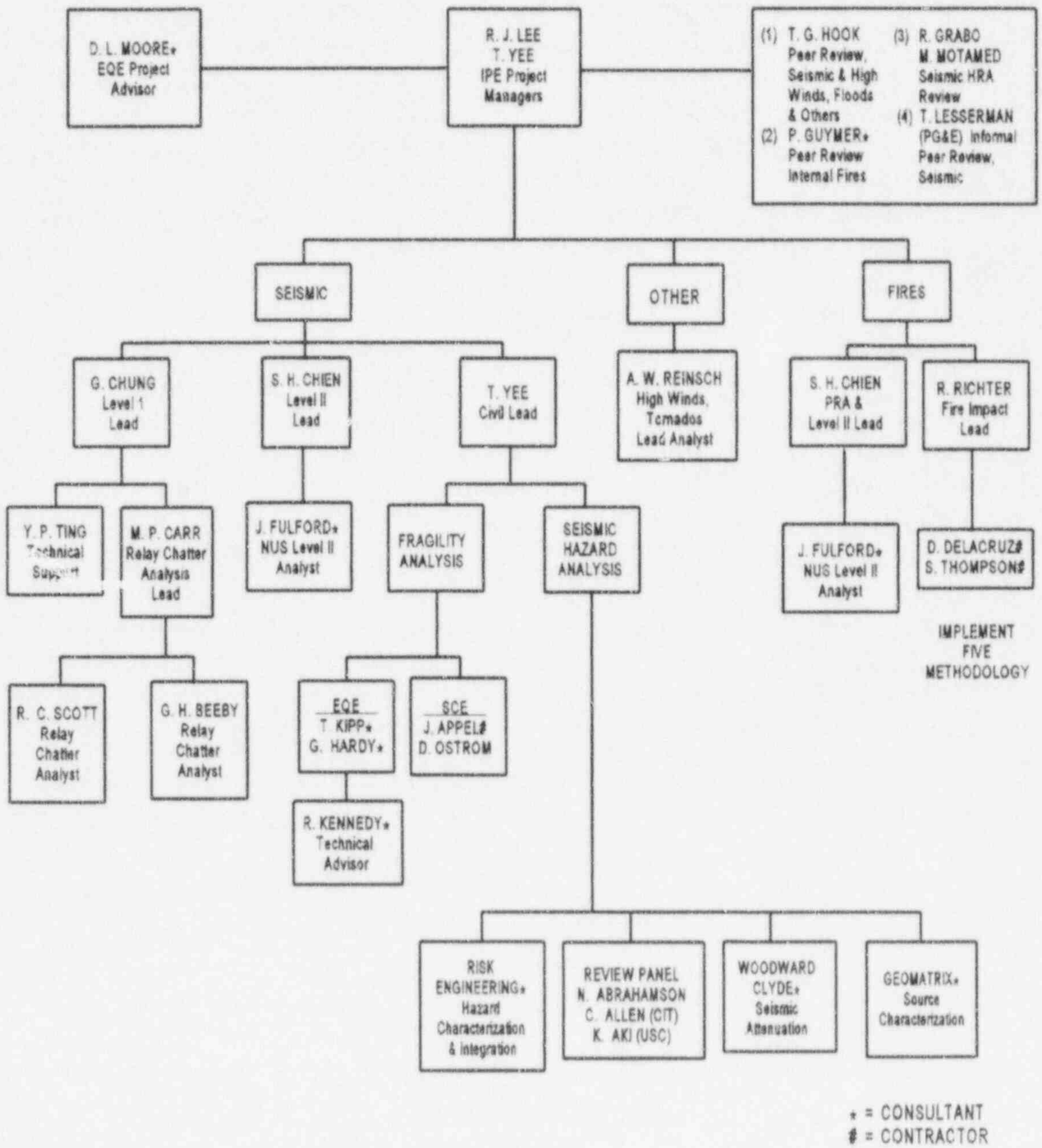
<u>Internal Fires</u>:

The internal fire analysis was a combined effort of the NSG and the NEDO-Fire Protection Group. The NSG performed quantification of conditional core damage probabilities (CCDPs), prepared human reliability assessments, integrated plant response, and completed the Level I & II quantification. NUS Corporation provided technical assistance for the Level II analysis. The NEDO-Fire Protection Group took the lead in the fire impact assessment which included implementation of EPRI developed Fire Induced Vulnerability Evaluation (FIVE) methodology, performed plant walkdowns, assessed non-Appendix R plant components, developed COMPBRN models for critical plant locations, and provided inputs for CCDP calculation.

High Winds, Floods, and Others

The Nuclear Safety Group completed the high winds, floods, and others risk analysis with technical guidance provided by EQE.

FIGURE 6.1-1
SONGS 2/3 IPEEE PROJECT ORGANIZATION CHART



6.2 COMPOSITION OF INDEPENDENT REVIEW TEAM

Several peer reviewers independently reviewed the seismic, fire, and high winds analysis (Table 6.2-1). The peer review of the seismic hazards characterization was performed by an independent panel of experts: Dr. Norman Abrahamson (consultant), Professor Keiiti Aki (USC) and Professor Clarence Allen (California Institute of Technology). The seismic equipment list was reviewed by various organizations within the NEDO. Seismic walkdown and fragility evaluations were reviewed by SCE and EQE. Mr. Robert Kennedy (consultant) provided a focused review of structure response analyses and fragility products. The seismic relay list and relay chatter evaluations were reviewed by the NEDO-Electrical Group. The human reliability assessments were reviewed by a senior reactor operator (Mr. Roger Grabo) in the Nuclear Training Department and by an independent engineer (Dr. Majid Motamed) of the Nuclear Safety Group who has been trained in the ASEP human reliability analysis methodology [6-1]. Mr. Thomas Hook of the PRA Group provided peer review for all seismic related aspects other than the seismic hazards characterization. The Assessment Engineering Group of the Nuclear Oversight Division (Mr. Jim Thomas) performed an independent review of selected portions of work conducted by NSG, NEDO and EQE. PG&E (Mr. Thomas Leserman) performed an informal, independent, limited scope review. Other than peer review, all those identified above did not otherwise materially participate in any aspect of the IPEEE effort.

Peer review of the internal fire analysis was performed by the Assessment Engineering Group (Mr. Jim Thomas) of the Nuclear Oversight Division, NUS Corporation (Mr. Paul Guymer) and Vectra Technologies, Inc. (Mr. J. Amason). Peer review of the high winds, floods and other hazards analysis was performed by Mr. Thomas Hook.

TABLE 6.2-1

LIST OF PEER REVIEWERS

Seismic Hazard:	Dr. Norman Abrahamson (Consultant) Professor Clarence Allen (CIT) Professor Keiiti Aki (USC)
Seismic Analysis:	Mr. Thomas Hook [SCE-Nuclear Safety Group (PRA subgroup)] Mr. Thomas Leserman (PG&E - PRA Supervisor)
HRA:	Mr. Roger Grabo (SCE-Training), Dr. Majid Motamed (SCE-NSG)

Fire Analysis: Mr. J. Amason (Vectra), Mr. Paul Guymer (NUS Corporation)

High Winds, etc.: Mr. Thomas Hook [SCE-Nuclear Safety Group (PRA subgroup)]

6.3 AREAS OF PEER REVIEW, COMMENTS & RESOLUTIONS

6.3.1 SEISMIC PEER REVIEW

6.3.1.1 Seismic Hazard Review

The expert review panel for the seismic hazard analysis was composed of the following individuals who are familiar with the seismicity of southern California:

Dr. Norman Abrahamson -- Consultant
Professor Keiiti Aki -- University of Southern California
Professor Clarence Allen -- California Institute of Technology

The panel provided an overview of the development of the San Onofre seismic hazard in the general areas of source attenuation, geology and seismology. Areas of review included the selection of seismic sources and their characterization, historical seismicity, ground motion attenuations and considerations for the experience gained from the Northridge Earthquake. Specific comments were raised about the possibility of blind thrust sources near San Onofre; weighting of ground motion attenuation equations; slip rates, rupture lengths and segmentation of faults; the observed seismicity in southern California and the prediction based on the characteristic magnitude distribution; and the correctness of the vertical spectra.

The review panel agreed with the methodology and results of the seismic hazard analysis conducted for SONGS. The comments of the review panel were resolved and incorporated into the SONGS seismic hazard final report [3-1].

6.3.1.2 Seismic Risk Analysis

The peer review for all other aspects of the seismic analysis (i.e., other than the seismic hazards characterization) focused on the following areas:

- conformance with the reporting guidance in NUREG-1407,
- development of the seismic event tree,

- the use of the IPE models in the quantification of seismic risk,
- the methodology for quantification of the seismic risk,
- review of the sequence cutsets,
- the consideration of correlation for seismic failures,
- technical issues identified in NUREG-1407 which may be applicable to San Onofre (e.g., performance of sensitivity studies for large uncertainties in the seismic hazard curve, potential for containment personnel hatch seal failure, potential for containment isolation valve failure due to relay chatter),
- significant differences between the Diablo Canyon and San Onofre seismic results, and
- the potential for multiple common cause breaks in the reactor coolant system to lead to a medium break loss-of-coolant accident (LOCA).

The major conclusions of the seismic peer review (other than the seismic hazards characterization) were:

- The seismic portion of the submittal conforms with the reporting guidance in NUREG-1407.
- The study methodology represents the state-of-the-art in seismic analysis.
- The quantification process was detailed, traceable, and performed with qualified codes.
- The correlation of seismic failures for like components was included in the analysis.
- Most of the significant differences between the Diablo Canyon and San Onofre seismic results can be attributed to differences in site soil geology. San Onofre is built on deep, well graded, dense sandy soils. Diablo Canyon is built directly on bedrock. The sandy soils at San Onofre result in a soil site spectra with a peak in the 2 to 5 Hz frequency range and dropping off toward the peak ground acceleration at the natural frequencies of most plant equipment. The rock site at Diablo Canyon results in a site spectra with a peak near the natural frequencies of most plant equipment. Therefore, at Diablo Canyon, the earthquake energy

in the amplified range of ground motion coincides with the natural frequencies of most plant equipment. At San Onofre, the earthquake energy is not significantly amplified in the range of ground motion coinciding with the natural frequencies of most equipment.

As an example of the differences attributable to the soil geology, the Diablo Canyon IPEEE assumed that the charging system was required to provide RCS makeup for all seismic events (i.e., RCS leakage would occur in all seismic events). This assumption is not applicable at San Onofre due to the ruggedness of the RCS (i.e., piping and tubing would not fail due to inertial forces or differential movements up to and beyond 4g). Walkdowns of the San Onofre RCS were performed to address the potential for non-safety related components falling on RCS piping in seismic events.

- There are only 10 small instrument lines connected to the RCS which could have a fragility below 8g. Multiple spatial interactions must occur otherwise to fail these lines. No such seismic-induced spatial interactions could be identified during the walkdown. In the event all 10 lines fail, the emergency core cooling system success criteria would not change from that assumed for a small LOCA.

The informal peer review by PG&E found that the report meets the intent of Generic Letter 88-20, Supplement 4 and NUREG-1407. PG&E noted that the methods and results appeared to be reasonably consistent with the Diablo Canyon Seismic PRA. All comments were either typographical or required minor clarification. All comments have been incorporated in this submittal.

6.3.2 INTERNAL FIRE PEER REVIEW

As with the seismic IPEEE, the fire IPEEE was reviewed by a number of organizations and individuals at various stages of the project. Internal reviews were performed by engineers working on the fire IPEEE as the calculations and documentation were developed. Independent reviews were also performed with Mr. J. Amason of Vectra reviewing the Phase II/III calculations and Mr. Paul Guymer of NUS reviewing all of the fire IPEEE documents. The peer review for the fire assessment included:

- verification that the methodology and data input were consistent with that required to satisfy the IPEEE requirements,
- systematic identification of fire-induced trip initiators,

- Phase I screening of assumptions regarding fire-induced near term shutdowns,
- fire boundaries and engineering evaluations to meet FIVE criteria,
- documentation of the offsite power cable identification analysis,
- evaluation of main feedwater components and cable locations,
- calculation of fire initiating event frequencies,
- selection of parameters and targets for the simplified Phase II fire modeling,
- treatment of high-low pressure interfaces and fire-induced interfacing systems LOCA,
- modeling of fire boundaries and thermal properties for the COMPBRN analysis,
- documentation of conditional core damage probability calculations,
- evaluation of dependencies between operator actions,
- documentation of all input data, assumptions, and engineering judgements, and
- verification of the software used for the fire IPEEE.

The comments of the two independent reviewers were considered and the analysis, calculations, and documentation were updated. For the particular items listed above, the following resolutions were performed:

- the calculations and tier 1 documentation were revised to conform to NUREG-1407 and FIVE guidance,
- a special analysis was performed to verify that the fire-induced trip initiators analysis and the Phase I screening was justifiable,
- documentation of the fire area boundaries and engineering evaluations was revised to meet reviewer comments,
- additional documentation and references were provided to document the offsite power cable identification,

- main feedwater cable and component locations were reviewed to ensure proper components were modeled,
- fire initiating event frequencies were recalculated or justified to meet reviewer comments,
- additional documentation as provided for the simplified Phase II fire modeling,
- a comprehensive analysis was performed to document plant protection against fire-induced interfacing systems LOCAs,
- a number of minor corrections were made to the detailed COMPBRN analyses, and additional documentation was provided,
- the tier 2 calculation packages are being reorganized to improve the traceability of the analyses,
- the fire scenarios and conditional core damage probability calculations were reviewed to verify that operator action dependencies were properly included in the calculations,
- a special effort was made to revise the tier 2 documentation to address all of the reviewer comments on assumptions and engineering judgments, and
- the COMPBRN and REBECA software have been verified using SCE procedures.

6.3.3 HIGH WINDS, FLOODS, AND OTHER EXTERNAL HAZARDS PEER REVIEW

The peer review of the high winds, floods, and other external events focused on completeness and accuracy of the analysis documentation. The primary effort in the evaluation was to ensure that the assumptions in the original (or updated) analyses remained applicable based on the current plant design and external factors impacting risk (e.g., weather, nearby transportation levels, etc.). The peer review of the high winds, floods, and other external events analysis did not identify any issues or discrepancies. The documentation and consideration of current design and external site features were complete.

6.4 REFERENCE

6-1 "Accident Sequence Evaluation Program (ASEP)," NUREG/CR-4772, February 1987.

7. PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

7.1 PLANT IMPROVEMENTS

Several plant and procedural modifications were identified during the course of the examination that provide various levels of risk improvement. These modifications are further explained below.

Seismic:

Plant improvement measures were identified during the walkdowns and during the calculation of the core damage sequences. During the walkdowns, the seismic capability engineers reviewed equipment on the seismic equipment list (SEL) to assess fragility and potential spatial interactions with other plant components and hazards. The description and implementation status of each plant improvement and procedural modification is provided below:

- improvement in the reliability of cross-connecting Units 2 and 3 to allow a unit's emergency diesel generator to supply power to the other unit in the event the other unit has a station blackout (improved 4kV power availability)

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- strengthening of ammonia tank supports (removes ammonia spill hazard)

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- removal of floor grating surrounding AFW valve actuators (allow valve movement without spatial interaction with surrounding grating)

  STATUS: Implementation by December 30, 1995.

- removal of concrete plug surrounding Unit 2 diesel generator fuel oil transfer piping (2) (improves piping's seismic capacity)

  STATUS: Implementation by December 30, 1995.

- fastening adjacent electrical cabinets/panels together (prevent interactions and relay chatter)

  STATUS: Implementation by March 31, 1996.

- stabilizing two light fixtures that interact with electrical cabinets

  STATUS: Implementation by December 30, 1995.

Fire:

The fire IPEEE was performed using the SONGS 2 Cycle 7 plant configuration, with the following future upgrades included in the assessment for the detailed Phase III analysis.

- For fire compartment 2-AC-30-20A (control room and cabinet area), implementation of an administrative change to Procedure SO23-13-2 (Shutdown from Outside the Control Room) would allow operators to utilize offsite power in the event that the reserve auxiliary transformers are not inadvertently tripped by fire induced damage to panel 2/3CR-63.

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- For fire compartments 2-DG-30-155 and 2-DG-30-158 (diesel generator rooms), implementation of an administrative change to Procedure SO23-13-21 (Fire) would allow operators to recover power to the 4 kV switchgear by disconnecting power to the diesel generator feeder breaker and reclosing the offsite power breaker on the switchgear.

  STATUS: Implementation by the end of the Cycle 9 refueling outage.

- For fire compartments 2-AC-50-44, 2-AC-50-45, 2-AC-50-46, 2-AC-50-47 (distribution rooms), 2-AC-50-35 and 2-AC-50-40 (switchgear rooms), implementation of an administrative change to alarm response procedure SO23-15-60.A1 (Annunciator Panel 60A, Emergency HVAC) would allow operators to use air duct and gas driven fans to prevent room heat-up.

  STATUS: Implemented.

7.2 UNIQUE SAFETY FEATURES

Several unique, plant-specific safety features help SONGS 2/3 to reduce plant risk against seismic, internal fire, and high winds, floods and other hazards. The unique features are listed below under each initiator.

SEISMIC

- SONGS 2/3 is designed for a very high design basis earthquake (.67g PGA) leading to very rugged design and construction. Structures, equipment, piping, cabinets and tanks are, in general, very strong, well-anchored, and braced.
- SONGS 2/3 is designed and constructed to modern standards and criteria.
- Potential spatial and seismic category II/I issues were addressed during the design of SONGS 2/3 to avoid system interactions.
- Portions of the fire suppression system are designed to seismic category I. Also, fire trucks and tankers are seismically restrained in an open area so as to be available following a seismic event.
- The risk from piping failure of hazardous material is very low since there are very few hydrogen or waste gas lines routed in areas with safety-related equipment. In addition, these lines are very strong and rugged.

INTERNAL FIRES

- The RCP seals have low vulnerability to seal LOCAs given loss of seal cooling. Therefore, fires causing loss of seal cooling would rarely result in a seal LOCA.
- Shared systems between units (offsite power, SWC, CCW) provide extra redundancy during fire events.
- SONGS 2/3 is a modern plant with generally good separation of redundant safety systems and cables.

OTHER EXTERNAL EVENTS

- SONGS 2/3 is located in an area with mild weather and environmental conditions, so there are relatively low outside impacts on the plant from weather-related external events

8. SUMMARY AND CONCLUSIONS

8.1 IPEEE SUMMARY RESULTS

Level I Results

The total mean core damage frequency (CDF) for *external* events at SONGS 2/3 was calculated to be approximately 3.3E-5/year[1]. (The CDF for *internal* event initiators [loss of coolant accidents (LOCAs), steam generator tube ruptures (SGTRs), loss of offsite power (LOP), etc.] was reported in the SONGS 2/3 IPE to be 3.0E-5/year.)

The results of the high winds, floods, and other hazards events analysis verifies that SONGS 2/3 meets the 1975 Standard Review Plan criteria, and that the analyzed and screened events do not pose a significant risk. In accordance with NUREG-1407, this study demonstrates that "no other plant-unique external event is known that poses any significant threat of severe accident within the context of the screening approach for 'High Winds, Floods and Others'".

Level II Results

No containment vulnerabilities were identified for the seismic, fire, and high winds, floods and other hazards analysis. Fission product release categories associated with containment bypass sequences were bounded by Level II results of the internal IPE.

8.2 RESOLUTION OF USIs, GIs, AND OTHER SAFETY ISSUES

The following NRC safety issues have been addressed in this IPEEE submittal:

SEISMIC ISSUES:

1.	USI A-45	Shutdown Decay Heat Removal Requirements-Seismic
2.	GI-131	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants
3.	USI A-40	Seismic Design Criteria
4.	USI A-46	Verification of Seismic Adequacy of Equipment in Operating Plants
5.	USI A-17	System Interactions in Nuclear Power Plants

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1 Following completion of all scheduled modifications discussed in Section 7.1.

6.	USI A-40	Seismic Capability of Large Safety-Related Above-Ground Tanks
7.	-----------	Eastern U.S. Seismicity (Charleston Earthquake) Issue
8.	GI-57	Effects of Fire Protection System Actuation on Safety Related Equipment-Seismic Issues
9.	-----------	Fire Risk Scoping Study-Seismic Issues

FIRE ISSUES:

1.	----------	Fire Risk Scoping Study-Fire Issues
2.	USI A-45	Shutdown Decay Heat Removal Requirements-Fire
3.	SECY-93-143	Reassessment of the NRC Fire Protection Program

Each of the above issues were discussed in the seismic or fire analysis sections given below, with the following summary resolution.

8.2.1 USI A-45: DECAY HEAT REMOVAL - SEISMIC

The seismic portion of the USI A-45 issue is discussed in section 3.8.1. In summary, a plant-specific systematic evaluation has been performed for SONGS 2/3 to identify any potential vulnerabilities in the decay heat removal systems. No vulnerabilities were identified for seismic initiating events. This issue is considered closed for SONGS 2/3.

8.2.2 GI-131: POTENTIAL SEISMIC INTERACTION INVOLVING THE MOVABLE IN-CORE FLUX MAPPING SYSTEM USED IN WESTINGHOUSE PLANTS

SONGS 2/3 is a Combustion Engineering design plant, and is not subject to this potential seismic interaction. Therefore, this issue is considered closed for SONGS 2/3.

8.2.3 USI A-40: SEISMIC DESIGN CRITERIA

USI A-40 is discussed in section 3.8.2. USI A-40 is not applicable to SONGS 2/3 since SONGS 2/3 are modern design plants, and the seismic design criteria addresses the issues identified in USI A-40. Therefore, this issue is considered closed for SONGS 2/3.

8.2.4 USI A-46: VERIFICATION OF SEISMIC ADEQUACY OF EQUIPMENT IN OPERATING PLANTS
USI A-17: SYSTEM INTERACTIONS IN NUCLEAR POWER PLANTS
USI A-40: SEISMIC CAPABILITY OF LARGE SAFETY-RELATED ABOVE-GROUND TANKS

The USI A-46 issue applies to older plants with a construction permit application docketed before 1972 and does not apply to SONGS 2/3. The scope of USI A-46 has been expanded by the NRC to include the seismic spatial system interaction of USI A-17 and the concern of USI A-40 for the seismic capability of large safety-related above-ground tanks. Spatial interactions were specifically addressed in the seismic capacity walkdowns and checklists, and the large safety-related yard tanks were demonstrated to have high seismic capacity. Therefore, all of these issues have been adequately addressed by the SONGS 2/3 seismic design criteria and methods and by the seismic capacity walkdowns. These issues are considered closed for SONGS 2/3.

8.2.5 EASTERN U.S. SEISMICITY (CHARLESTON EARTHQUAKE) ISSUE

This issue is not applicable to SONGS 2/3 and is considered closed.

8.2.6 GI-57: EFFECTS OF FIRE PROTECTION SYSTEM ACTUATION ON SAFETY RELATED EQUIPMENT, AND NUREG/CR-5088, FIRE RISK SCOPING STUDY-SEISMIC ISSUES

GI-57 and the Fire Risk Scoping Study (FRSS) raised questions regarding seismic-fire interactions, and the potential impacts of spurious actuation of fire protection systems. Seismic-induced fire/flood interaction issues, including spurious actuation of the fire protection systems, were evaluated in detail as discussed in section 3.3.4. The overall result is that any potential seismic-induced fires or floods will not affect safety equipment needed for shutdown during or after a seismic event. These issues are considered closed for SONGS 2/3.

8.2.7 FIRE RISK SCOPING STUDY-FIRE ISSUES

The Fire Risk Scoping Study issues related to fires are discussed in section 4.6. Based on this evaluation, the issues presented in the FRSS were found to have no unanalyzed impact on fire risk at SONGS 2/3 and are considered closed.

8.2.8 USI A-45: DECAY HEAT REMOVAL-FIRE

The fire portion of the USI A-45 issue is discussed in section 4.7.1. In summary, a plant-specific systematic evaluation has been performed for SONGS 2/3 to identify any potential vulnerabilities in the decay heat removal systems. No vulnerabilities were identified for fire initiating events. This issue is considered closed for SONGS 2/3.

8.2.9 SECY-93-143: REASSESSMENT OF THE NRC FIRE PROTECTION PROGRAM

The specific issues listed in Table 10.2-3 in SECY-93-143 to be included in the IPEEE are discussed in section 4.7.2. Based on this evaluation, these issues were found to have no unanalyzed impact or potential vulnerability for fire risk at SONGS 2/3 and are considered closed.

8.3 CONCLUSIONS

The objective of the SONGS 2/3 IPEEE, as outlined in Generic Letter 88-20, Supplement 4, was for SCE:

1) to develop an appreciation of severe accident behavior,

2) to understand the most likely severe accident sequences that could occur at SONGS 2/3 under full power conditions,

3) to gain a qualitative understanding of the overall likelihood of core damage and radioactive material release, and

4) if necessary, to reduce the overall likelihood of core damage and radioactive material releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

This examination documents compliance with the first three objectives where a large majority of the study was performed in-house. The examination tasks performed by SCE include plant modeling, walkdowns, Level 1 and 2 plant response analysis, and quantification of results. Completion of these tasks results in a quantitative and qualitative understanding of the likely severe accident sequences.

As a result of the examination, SCE identified several plant and procedural modifications which would provide substantive and cost-effective risk benefit. All modifications will be implemented no later than the end of the Cycle 9 refueling outage. This demonstrates that the SONGS 2/3 IPEEE meets the fourth objective.