WASHINGTON PUBLIC POWER
SUPPLY SYSTEM

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS
WASHINGTON NUCLEAR PLANT 2

MAIN REPORT

Revision 0: JUNE 1995

Washington Public Power Supply System
3000 George Washington Way
Richland, Washington 99352

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F. Steele	System Analysis & Relay Work

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ADS	AUTOMATIC DEPRESSURIZATION SYSTEM
ANSI	AMERICAN NATIONAL STANDARDS INSTITUTE
ARI	ALTERNATE ROD INSERTION SYSTEM
ASEP	ACCIDENT SEQUENCE EVALUATION PROGRAM
ASME	AMERICAN SOCIETY OF MECHANICAL ENGINEERS
ATWS	ANTICIPATED TRANSIENT WITHOUT SCRAM
BHEP	BASIC HUMAN ERROR PROBABILITY
BPA	BONNEVILLE POWER ADMINISTRATION
BWR	BOILING WATER REACTOR
BWROG	BOILING WATER REACTOR OWNERS GROUP
CAS	CONTROL AND SERVICE AIR SYSTEM
CCF	COMMON CAUSE FAILURE
CCFP	CONDITIONAL CONTAINMENT FAILURE PROBABILITY
CDF	CORE DAMAGE FREQUENCY
CEP	CONTAINMENT EXHAUST/PURGE SYSTEM
CET	CONTAINMENT EVENT TREE
CFAR	COMPONENT FAILURE ANALYSIS REPORT
CHR	CONTAINMENT HEAT REMOVAL
CIA	PRIMARY CONTAINMENT INSTRUMENT AIR
CN	CONTAINMENT NITROGEN SYSTEM
COND	MAIN CONDENSATE SYSTEM
CPI	CONTAINMENT PERFORMANCE IMPROVEMENT
CRD	CONTROL ROD DRIVE SYSTEM
CST	CONDENSATE STORAGE TANK
CSTS	CONDENSATE STORAGE AND TRANSFER SYSTEM
CW	CIRCULATING WATER SYSTEM
DCH	DIRECT CONTAINMENT HEATING
DEH	DIGITAL ELECTRO-HYDRAULIC CONTROL SYSTEM
DHR	DECAY HEAT REMOVAL
ECCS	EMERGENCY CORE COOLING SYSTEM
ECOM	ERROR OF COMMISSION
EDG	EMERGENCY DIESEL GENERATOR
EDR	EQUIPMENT DRAIN PROCESSING
EOM	ERROR OF OMISSION
EOP	EMERGENCY OPERATING PROCEDURE
EPG	EMERGENCY PROCEDURE GUIDELINES
EPRI	ELECTRIC POWER RESEARCH INSTITUTE
ESF	ENGINEERED SAFETY FEATURE
FCI	FUEL-COOLANT INTERACTIONS
FDR	FLOOR DRAIN PROCESSING
FP	FIRE PROTECTION SYSTEM
FSAR	FINAL SAFETY ANALYSIS REPORT
FVI	FUSSEL-VESSELLY IMPORTANCE
GSI	GENERIC SAFETY ISSUES

TABLE OF ACROYNMS (Cont'd)

HPCS	HIGH PRESSURE CORE SPRAY SYSTEM
HPME	HIGH PRESSURE MELT EJECTION
HRA	HUMAN RELIABILITY ANALYSIS
HVAC	HEATING, VENTILATION, AND AIR CONDITIONING
IDCOR	INDUSTRY DEGRADED CORE RULEMAKING GROUP
INPO	INSTITUTE OF NUCLEAR POWER OPERATIONS
IORV	INADVERTANTLY OPENED RELIEF VALVE
IPE	INDIVIDUAL PLANT EXAMINATION
IPEEE	INDIVIDUAL PLANT EXAMINATION EXTERNAL EVENTS
IPEM	INDIVIDUAL PLANT EXAMINATION METHODOLOGY
IPEP	INDIVIDUAL PLANT EXAMINATION PARTNERSHIP
IREP	INTERIM RELIABILITY EVALUATION PROGRAM
ISLOCA	INTERFACING SYSTEM LOSS OF COOLANT ACCIDENT
LER	LICENSEE EVENT REPORT
LFMG	LOW FREQUENCY MOTOR GENERATORS
LOCA	LOSS OF COOLANT ACCIDENT
LOOP	LOSS OF OFFSITE POWER
LPCI	LOW PRESSURE COOLANT INJECTION
LPCS	LOW PRESSURE CORE SPRAY SYSTEM
LPME	LOW PRESSURE MELT EJECTION
MCC	MOTOR CONTROL CENTER
MCCI	MOLTEN CORE-CONCRETE INTERACTIONS
MOV	MOTOR OPERATED VALVE
MSIV	MAIN STEAM ISOLATION VALVE
NCR	NONCONFORMANCE REPORT
NEI	NATIONAL ENERGY INSTITUTE
NFPA	NATIONAL FIRE PROTECTION ASSOCIATION
NPRDS	NUCLEAR PLANT RELIABILITY DATA SYSTEMS
NPSH	NET POSITIVE SUCTION HEAD
NRC	NUCLEAR REGULATORY COMMISSION
NREP	NATIONAL RELIABILITY EVALUATION PROGRAM
NSAC	NUCLEAR SAFETY ANALYSIS CENTER
NS4	NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM
NUMARC	NUCLEAR MANAGEMENT AND RESOURCES COUNCIL
PCS	POWER CONVERSION SYSTEM
PDS	PLANT DAMAGE STATE
PP	POWER PANEL
PPM	PLANT PROCEDURES MANUAL
PRA	PROBABALISTIC RISK ASSESSMENT
PSA	PROBABILISTIC SAFETY ASSESSMENT
RCC	CLOSED COOLING WATER SYSTEM
RCIC	REACTOR CORE ISOLATION COOLING
RFW	REACTOR FEEDWATER SYSTEM
RHR	RESIDUAL HEAT REMOVAL SYSTEM

TABLE OF ACROYNMS (Cont'd)

RHVAC	REACTOR BUILDING EMERGENCY COOLING SYSTEM
RPS	REACTOR PROTECTION SYSTEM
RPV	REACTOR PRESSURE VESSEL
RRC	REACTOR RECIRCULATION COOLING SYSTEM
RRW	RISK REDUCTION WORTH
RWCU	REACTOR WATER CLEANUP SYSTEM
SBO	STATION BLACKOUT
SCRAM	SAFETY CONTROL ROD AXE MAN
SDV	SCRAM DISCHARGE VOLUME
SGT	STANDBY GAS TREATMENT
SJAE	STEAM JET AIR EJECTOR
SLC	STANDBY LIQUID CONTROL SYSTEM
SMS	SCHEDULED MAINTENANCE SYSTEM
SORV	STUCK OPEN RELIEF VALVE
SOV	SOLENOID OPERATED VALVE
SP	SUPPRESSION POOL
SPC	SUPPRESSION POOL COOLING
SRV	SAFETY RELIEF VALVE
SSEL	SAFE SHUTDOWN EQUIPMENT LIST
STG	SOURCE TERM GROUP
SW	STANDBY SERVICE WATER SYSTEM
TAF	TOP OF ACTIVE FUEL
TMI-2	THREE MILE ISLAND UNIT 2
TSW	PLANT (TURBINE) SERVICE WATER
USI	UNRESOLVED SAFETY ISSUES
WNP-2	WPPSS NUCLEAR PROJECT 2
WPPSS	WASHINGTON PUBLIC POWER SUPPLY SYSTEM

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1.0 EXECUTIVE SUMMARY

1.1 Background and Objectives

This report presents the Individual Plant Examination of External Events (IPEEE) information for Washington Nuclear Plant-2 (WNP-2) requested by the NRC in Generic Letter 88-20, Supplement 4 (Reference 1.5.1). Its content and format are prepared in conformance to NUREG-1407 (Reference 1.5.2). WNP-2 is a General Electric BWR 5 boiling water reactor with a Mark II containment operated by the Washington Public Power Supply System (The Supply System). It is located on the DOE Hanford Site near Richland, Washington.

The WNP-2 IPEEE represents a systematic evaluation using conventional probabilistic safety assessment (PSA) methodology to identify potential vulnerabilities to severe accidents. In performing this plant specific examination the objectives stated in the generic letter were achieved, namely:

- to develop an appreciation of severe accident behavior,
- to understand the most likely severe accident sequences that could occur at the plant under full power conditions,
- to gain a more qualitative understanding of the overall probabilities of core damage and fission product releases and,
- if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

During the performance of the IPEEE for WNP-2, the Supply System maximized its use of in-house personnel. Experts from outside the corporation were used to train and assist Supply System staff in the general application of PSA techniques with respect to external events. In addition, Experts from outside the company were used to perform many of the tasks requiring a deep technical background in specific subjects; most notably the soil structure interaction issues and component fragility analysis. Supply System personnel performed the following tasks:

- compilation of the safe shutdown equipment lists for seismic and fire PSA models,
- relay chatter screening evaluation of electric relays, contactors and switches,
- PSA model quantification for seismic and fire PSA,
- technical review of the analysis,

- formal in-house independent review to validate the IPEEE process and results, and
- participation in plant walkdowns conducted for the external events evaluations.

1.2 Plant Familiarization

The Washington Nuclear Plant 2 (WNP-2) is a boiling water reactor of General Electric BWR 5 design. The reactor rated thermal power is 3323 MWt with a design electrical output of 1154 MWe. The Construction Permit was granted in March of 1973 and the plant began commercial operation in December 1984. Commencing with Cycle 11 operation in June 1995, the reactor rated power will be 3486 MWt with a design electrical output of 1230 MWe. An evaluation of the power uprate impact on the WNP-2 IPE was conducted by General Electric and the Supply System. It was concluded that the power uprate did not impact the WNP-2 core damage frequency.

WNP-2 is located in the Hanford Reservation approximately 10 miles north of Richland and 3 miles west of the Columbia river. Approximately 135 miles to the west-southwest of the site is Mount St. Helens, an active volcano. The plant site is in a continental-type climate; as a result, the site experiences wide variations in temperature conditions. Historical data reveal a temperature range of -27°F to 115°F. The maximum recorded precipitation for a single month was 3.08 inches and the maximum recorded yearly precipitation was 11.45 inches.

The reactor is housed in a free standing steel primary containment of a Mark II design. The free standing steel design is unique for domestic Mark II containments. The steel construction results in a decreased propensity for catastrophic failure modes, but has a lower ultimate pressure capacity than a containment with a steel lined concrete design. With this design, the drywell which houses the reactor vessel is placed over the wetwell volume. The drywell to wetwell connection is through 99 downcomers and the drywell concrete floor is sealed to containment at its periphery with a metal Omega seal. The reactor sits on a pedestal that is recessed below the drywell floor area and is founded on the wetwell floor. The containment vessel is founded on the reactor building basemat. During power operation, the containment is inerted with a nitrogen atmosphere.

The majority of the safety related equipment as well as the containment vessel is housed in the secondary containment, also referred to as the reactor building. Subatmospheric pressure is maintained within this building to prevent the leakage of radioisotopes. At or below the refueling floor, the reactor building is constructed of reinforced concrete, with lateral seismic loads resisted by shear walls and floor slabs. The refueling floor, at the top of the building, is enclosed by a steel superstructure.

The majority of the equipment required to convert heat energy to electrical energy is housed in the turbine generator building. The lateral load resisting system for the turbine building consists of reinforced concrete shear walls and floor diaphragms. The operating floor of the turbine generator building is enclosed by a steel superstructure. The Turbine Generator building is located adjacent to the reactor and radwaste/control buildings.

The main and remote shutdown control rooms, the safety related switch gear rooms, the battery rooms, and the waste processing equipment is located in the radwaste/control building. The radwaste/control building is primarily a reinforced concrete structure, with lateral loads resisted by shear walls and floor diaphragms. The radwaste/control building is located adjacent to the reactor and turbine generator buildings.

The emergency diesel generators are located in their own building adjacent to the reactor and radwaste buildings. The diesel generator building is a reinforced concrete structure, with lateral loads resisted by shear walls and floor diaphragms.

WNP-2 has two separate divisions of standby service water. Each division of pumping equipment is located in its own building both of which are separated from the other plant buildings. The service water pump houses are reinforced concrete structures, with lateral loads resisted by shear walls.

The plant design includes redundant and diverse systems, most of which have automatic initiation, to maintain all critical safety functions when the plant is being brought to a safe shutdown condition. These systems include:

- reactivity control by CRD/RPS, backed up by ARI and SLC,

- reactor pressure control by 18 SRVs, which have a combined energy relieving capacity of greater than 100% normal reactor power,

- three high pressure coolant injection systems:

  RFW, HPCS, and RCIC, (CRD flow is available but has insufficient flow capacity to justify its being credited for core cooling by this analysis),

- automatic depressurization, with redundant initiation logic and seven ADS valves,

- low pressure coolant injection systems: COND, LPCS, LPCI, SW-Crosstie, and Fire Protection (FP) Water,

- three systems for containment heat removal: RHR, Power Conversion System (PCS), and CEP (Venting).

  NOTE: The term PCS is intended to represent the secondary system heat removal function provided by the feedwater, condensate and condenser systems, and their associated support systems.

WNP-2 has two independent offsite power sources and two independent divisions of emergency AC power which are capable of providing all of the electrical power needed to bring the plant to safe shutdown condition. A third independent source of emergency AC power is available to power one division of high pressure injection and a third offsite power source can be made available within several hours if manual hookup and initiation is successful.

The fire protection system at WNP-2 consists of a fire detection and alarm system and four different types of fire suppression systems as follows: Water, carbon dioxide, control room halon 1301 floor module suppression system, and portable equipment.

Revision 4 of the BWR Owners Group EPG was used to develop WNP-2's symptom-based emergency operating procedures and WNP-2's operator training program is accredited by INPO.

1.3 Overall Methodology

The WNP-2 IPEEE was performed in accordance with the general precepts and methods outlined in the PSA procedures as described in NUREG-1407, and the guidelines presented in Generic Letter 88-20, its supplements and appendices.

The external events of internal fire and earthquake were addressed using PSA methodology. The external events PSA for fire and seismic were performed using Revision 1 of the WNP-2 IPE (Reference 1.5.3), the most recently updated risk assessment at WNP-2. Revision 1 to the IPE incorporates changes in plant design and procedures that have occurred since the design was originally defined for Revision 0 of the IPE on the basis of a 1989 "freeze date." The current revision (dated July 1994) uses plant data and system design as it was at the end of 1993. In addition, Revision 1 to the IPE also reflects enhancements which better describe the plant's response to issues of SRV reliability, Loss of Offsite Power and plant specific initiators.

The WNP-2 IPE uses the fault tree linking approach to identify and quantify individual accident sequences. With this approach, functionally descriptive event trees are used to model the possible accident sequences which result in core damage, and initiator-specific containment event trees are used to represent possible containment behavior following a core melt accident. Detailed system fault trees are merged to quantify individual event tree sequences in both the Level 1 and Level 2 analyses, although the Level 2 system fault trees were much simpler. The NUPRA computer code was used to perform the quantification and both the RETRAN and MAAP computer codes were used to characterize plant, containment, and fission product behavior.

The effects of common cause failures are included as basic events in the WNP-2 system fault trees and are quantified by the Beta method. This method provides a numerical estimate for the common cause failure probability which should be assigned to account for the likelihood that the plant would experience simultaneous failures of two or more identical components in a single train or multiple trains in a single system.

The external events evaluation for seismic activity started with a seismic hazard analysis to estimate the frequency of various levels of seismic ground motion at the site. The structure and component fragility analyses were then performed. Screening to determine which structures and components required fragility curves was performed using a review level earthquake (RLE) of 0.5g, as requested for western plant sites (Generic Letter 88-20 Supplement 4). Three walkdowns were performed to inspect the plant equipment and mounting details, and to look for structure, system or component interaction issues. Those components which required fragility curves were incorporated in the WNP-2 PSA model to find the various combinations of structural and random failures that could propagate to a core damage accident sequence. The results of the core damage sequences and the seismic hazard analysis were then combined to yield estimates of seismic risk using the computer code EQESRA.

The external events evaluation for internal fires was performed with PSA technology but also utilized some portions of the FIVE methodology (Reference 1.5.4) for systematic screening and ignition source frequency determination. The internal fire analysis began by identifying and locating all equipment critical to plant safety and tracing the supporting electrical cable. Fire areas were identified, relying heavily on work that was performed for compliance with Appendix R requirements. A detailed walkdown of the plant fire areas was conducted to identify areas of vulnerability, confirm fire suppression system details, and identify combustibles and ignition sources. Seismic/fire interactions were also assessed during plant walkdowns. FIVE methodology was employed to screen fire areas and to determine ignition source frequencies. The COMPBRN IIIe computer code (Reference 1.5.5) was utilized to determine fire growth and spread characteristics in critical fire areas. Fire initiating events in each fire area and the resulting equipment damage was combined with random equipment failure modes using the WNP-2 PSA model to determine core damage frequency estimates.

The remaining external events i.e., volcano eruption, high wind, external flood, transportation or nearby facility accidents, severe temperature transients, severe storms, lightning, external fires, extraterrestrial activity and earth movement did not require PSA analysis. These events were assessed using the progressive screening approach recommended in Generic Letter 88-20. Since WNP-2 is a relatively new plant, the 1981 version of the Standard Review Plan (SRP) was used by the NRC and the Supply System to assess the safety design margin of the plant. Therefore, the 1981 version of the SRP was used for this IPEEE evaluation rather than the 1975 version as permitted by the generic letter.

1.4 Summary of Major Findings

The major conclusions of the IPEEE are summarized here. More detail is found in the sections dealing with each of these specific potential initiating events.

1.4.1 Seismic PSA Results

WNP-2 was very conservatively constructed for a 0.25g horizontal peak ground acceleration (PGA) and for the most part had very large margins such that most systems, structures, and components could be screened out from additional, more detailed, seismic analyses. Spatial systems interactions identified for additional potential assessment were a few cases of possible cabinet impacts which could only affect relay performance. No spray or flood issues were noted although there is one case where failure of a liquid nitrogen storage tank could possibly flood the diesel generator air inlets with gaseous nitrogen. No significant seismic/fire interactions were noted.

A new probabilistic soil-structure interaction analyses was performed in conjunction with the seismic IPEEE study which demonstrated a very large margin in the original safe shutdown earthquake (SSE) instructure spectra and structural loads. Based upon a comparison of structural loads and some supporting structural calculations, all structures were screened out. A comparison of SSE and new probabilistic spectra, comparisons of qualification margin relative to the SSE and a detailed plant walkdown enabled most equipment to be screened out. The exceptions to this were certain motor control centers and a few instrument and control cabinets. The motor control centers were not screened out in some of the higher demand building locations.

The overall impression from the walkdowns and the review of the seismic qualification documentation is that the plant is well constructed and has a high resistance to seismic loading.

The switchyard was found to have a low seismic capacity, typical of switchyards across the nation. Loss of offsite power (LOOP) sequences dominate the seismic core damage frequency analysis. The failure of critical switchgear room cooling sequence also is a major contributor at about 22% of the total seismic induced CDF.

The overall CDF from seismic events was calculated to be 2.1×10^{-5}/yr. MCC failures play a contributing role in most of the top sequences and one of the recommendations from the IPEEE analysis is to look at the cost effectiveness of strengthening the anchorage of the most important MCCs. This result is conservative because no recovery actions have been modeled.

TABLE 1.4-1

WNP-2 IPEEE Dominant Seismic Sequences

BRIEF SEQUENCE DESCRIPTION	SEQUENCE	% OF CDF
LOOP with DG-1 and DG-2 failure	E(E)S09	51.9%
Failure of critical switchgear room cooling	EMS09	21.8%
LOOP with failure to establish suppression pool cooling	E(E)S02	14.8%
LOOP with failure of all screened equipment	E(E)S10	6.6%

1.4.2 Fire PSA Results

For purposes of consistency, the division of the plant into compartments or fire zones are to a large extent those defined by the WNP-2 Fire Hazard Analysis developed in the WNP-2 Appendix R fire analysis. In a select number of instances where fire areas are very large, sub-compartments were established to reduce excessive conservatism where judgement, previous experience and/or fire damage simulations deemed them appropriate. The equipment hatches and stairwells were treated as separate fire areas to ensure propagation effects were accounted for in the analysis. The Control Room fire was analyzed separately.

A significant portion of the Fire PSA involved walkdowns to identify or confirm fire barriers and fire areas. No significant problems were noted during the walkdowns. There were several instances of cable trays being protected and not noted on the drawings. These were instances of where Appendix R analysis deemed the protection was not required and it was abandoned in place. No credit for these barriers was taken in the analysis.

The initial analyses assumed all equipment in a given fire area was destroyed and no credit for recovery or manual suppression was taken. After combining the fire initiation frequency with the conditional core damage frequency, the fire area was screened as being not significant if the calculated core damage frequency was less than 1E-06/yr. Sixteen fire areas remained after this conservative screening: Div 1 and Div 2 switch gear rooms, Div 1 and Div 2 electrical equipment rooms, Div 1 and Div 2 battery rooms, Remote Shutdown Room, one turbine generator building area, the turbine generator building corridor, the corridor between the electrical equipment and battery rooms, and five reactor building areas.

The fire areas remaining after the initial screening were requantified accounting for potential recovery actions and more realistic fire ignition frequencies. No credit was taken for manual suppression capability. The dominant fire areas and their contribution to core damage frequency are summarized in Table 1.4-2.

The dominant fire areas exhibit the same sequences leading to core damage; namely, the fire renders venting, power conversion system, and one train of RHR or SW unavailable, such that the other decay heat removal train unavailability dominates the sequences. Some of the Reactor Building fire area sequences exhibit a loss of high and low pressure injection capability which makes loss of decay heat removal immaterial.

The contribution to core damage frequency from the Fire PSA is 9.16E-06/yr. The Control Room fire contributes an additional 8.4E-06/yr to the core damage frequency.

TABLE 1.4-2

WNP-2 IPEEE Dominant Fire Areas

BRIEF FIRE AREA DESCRIPTION	FREQUENCY (1/yr)	% OF CDF
TG-1K, Turb Gen Corridor	2.91E-06	32%
RC-6, Div 2 Battery Room	1.48E-06	16%
RC-19, Div 1/2 Elec/Batt Rm Corridor	1.06E-06	12%
TG-1A, Turb Gen Bldg West 441′	5.91E-07	6%
RC-4, Div 1 Elec Equipment Room	5.54E-07	6%
R-1D, NW Reactor Building 471′	4.87E-07	5%
R-1M, Equipment Hatch	3.77E-07	4%
RC-7, Div 2 Elect Equipment Room	4.06E-07	4%
R-1J, Reactor Building 522′	2.91E-07	3%
R-1H, NW Reactor Building 501′	2.90E-07	3%
RC-5, Div 1 Battery Room	2.55E-07	3%
RC-9, Remote Shutdown Room	1.66E-07	2%
RC-8, SWGR Rm #2	9.43E-08	2%
R-1B, NE Reactor Building 471′	7.35E-08	1%
RC-14, SWGR Rm #1	1.29E-07	1%
R-10, Stairwell S-3	0.0	0%
TOTAL	9.16E-06	100%

1.4.3 Volcano Eruption Evaluation Results

After the 1980 eruptions of Mount St. Helens, the NRC and the United States Geological Survey (USGS) indicated that the FSAR estimates of the uncompacted thickness of ashfall and the rate of ashfall previously used by the Supply System were not conservative. The Supply System established a task force to evaluate and recommend a plan to incorporate the findings. The task force results were reported to the NRC in a letter dated October 4, 1982 (Reference 1.5.6). WNP-2 plant systems and equipment were evaluated with respect to operability and reliability during a design basis ashfall using new and more conservative values for maximum compacted and uncompacted ashfall thicknesses. In addition, the Supply System agreed to develop a warning system tied to the USGS warning system.

Using the results of the task force evaluation, a specific plant procedure was developed that utilizes appropriate equipment modifications and designated filters for safety related systems in nearby trailers to ensure safe operation and shutdown of the plant following a design basis ashfall.

Based on the task force assessment and the resulting plant procedure modifications including the addition of advanced warning provisions and equipment modifications, it is concluded that WNP-2 is assured of safe plant operation and shutdown following a volcanic event. Therefore, no further examination of this event will be necessary.

1.4.4 High Winds Evaluation Results

WNP-2 design criteria are for wind speeds of 300 mph rotational and 60 mph translational with a pressure drop of 3 psi occurring at 1.0 psi/sec. Compared to Regulatory Guide 1.76 criteria for the Western US region of 190 mph rotational and 50 mph translational with a pressure drop of 1.5 psi occurring at 0.6 psi/sec, the current requirements represent a robust design for WNP-2. The wind borne missiles selected for the design of WNP-2 structures and equipment are representative of the site and meet the requirements of Regulatory Guide 1.76, GDC 2, GDC 4, and Regulatory Guide 1.117. Thus, WNP-2's current tornado design criteria meet the requirements of the SRP, and no further evaluation will be necessary.

1.4.5 External Flood Evaluation Results

WNP-2 has been designed to withstand the effects of external flood resulting from either extraordinary precipitation or from flooding on the Columbia River. The buildings and site topology have been designed to withstand a probable maximum precipitation (PMP) event of 9.2 inches in a six hour period. This PMP value is extremely conservative when compared to the maximum recorded precipitation of 1.68 inches in six hours. The PMP value is higher than site specific historic rainfall would suggest because it was based on readings from two weather stations that are located west of the Cascade Mountain range. The roofs of safety related buildings, the diesel generator building, radwaste/control building, reactor building, and the service water pumphouses as well as the storm sewer system have been adequately designed for the PMP and meet the requirements of GDC 2. Thus, the WNP-2 plant design meets the requirements of appropriate sections of the SRP pertaining to this external event.

The Columbia River dam and reservoir system would significantly reduce any potential for flooding along the river. Estimation of the probable maximum flood in a manner consistent with Regulatory Guide 1.59, rev. 2 and the requirements of GDC 2 result in a flood level of 390 ft Mean Sea Level (MSL). This is 51 feet below the plant elevation of 441 ft MSL. Thus, floods due to extraordinary basin run off pose no threat to the plant. WNP-2 has also been evaluated against flooding caused by dam failures. The worst case scenario is a postulated failure of Grand Coulee Dam. This would conservatively result in a flood level of 424 ft MSL. Thus, for this postulated event the plant design reflects the requirements of GDC 2, and therefore, meets the SRP requirements. No further evaluation of external flooding is required.

1.4.6 Transportation and Nearby Facility Accident Evaluation Results

Hazards postulated to result from accidents during nearby transportation, industrial, and military activities including accidental release of on-site hazardous material were considered in this evaluation. Transportation activities considered included aviation, marine, pipeline, railroad and trucks carrying hazardous materials. The evaluation of nearby facility accidents considered all facilities within 5 miles of the plant. No military facilities are located within 5 miles of WNP-2. In all cases, SRP requirements were met. No further evaluation of these external events are required.

1.4.7 Other External Events Evaluation Results

The external events evaluation also considered severe temperature transients, severe storms (ice, hail, snow, and sand), lightning strike, external fires, extraterrestrial activity, and earth movement. None of these events were found to pose a significant threat to the safe operation of WNP-2.

1.5 References

1.5.1 U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE)," Generic Letter No. 88-20, Supplement 4, June 1991

1.5.2 U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE)," NUREG-1407, June 1991

1.5.3 Supply System, "Individual Plant Examination Washington Nuclear Plant 2," WPPSS-FTS-133, July 1994

1.5.4 EPRI, "Fire-Induced Vulnerability Evaluation (FIVE) Methodology Plant Screening Guide," TR-100370, December 1991

1.5.5 University of Los Angeles, "COMPBRN IIIe: An Interactive Computer Code for Fire Risk Analysis," EPRI-NP-7282, Project 3000-39, May 1991

2.0 EXAMINATION DESCRIPTION

2.1 Introduction

In response to NRC Generic Letter No. 88-20, Supplement 4, the Supply System has performed an Individual Plant Examination of External Events of WNP-2 for possible severe accident vulnerabilities. Consistent with the general purpose of the generic letter, the Supply System has utilized in-house technical staff from various departments (Safety Analysis, Plant Technical, Engineering, Operations, and Licensing) to perform or participate in the IPEEE work. Consultants were utilized for assistance in data gathering, for peer review and comments on all aspects of the IPEEE to ensure industry acceptable methods were used and a thorough examination was performed. In addition, outside experts were utilized for soil structure interaction analysis, building structure fragility and component fragility analysis.

The external analysis was performed in three parts: Seismic probabilistic safety analysis with assistance from TENERA and EQE International; Fire probabilistic safety analysis with assistance from VECTRA and Haliburton NUS; and the remaining external events evaluation with assistance from TENERA.

2.2 Conformance with Generic Letter and Supporting Material

This IPEEE report conforms to the NRC guidance given in Generic Letter 88-20, Supplement No. 4 (Reference 1.5.1), and NUREG-1407 (Reference 1.5.2). This report's section numbers and titles are based on the outline given in Table C.1 of NUREG-1407. Documentation of examination details and results (referred to as Tier 2 documentation herein) are kept in a traceable manner under in-house document control as required by Section 10 of the Generic Letter. This IPEEE report contains all of the information required by the submittal guidance document, NUREG-1407. Since the IPEEE represents recently completed analysis, this report reflects the current designs and practices at WNP-2 as of December 1993.

2.3 General Methodology

The general methodology used is external event specific. However, both the seismic examination and the internal fire examination utilize PSA methods and are based on revision 1 of the WNP-2 IPE as submitted to the NRC in July of 1994 (Reference 1.5.3). The system analysis methodology of the WNP-2 PSA utilized the fault tree linking approach. The event trees combined the initiating event with system functional successes or failures to delineate the accident sequences.

The Supply System developed very detailed fault trees for the IPE submittal, taking into account component failures, initiation and control failures (including logics and interlocks), support system failures, test and maintenance unavailabilities, operator errors, and common cause failures. Fault trees were developed down to the component, relay, and sensor levels beyond which component failure rate data do not exist. These fault trees were modified by inserting house events which allow components and/or systems to be disabled in the analysis to account for seismic or internal fire damage.

The Supply System developed several engineering standards used in developing the event and fault trees and quantifying the results. The convention for defining basic faults conforms to the WNP-2 Master Equipment List which has an unique ID for each component. The standards, together with configuration control of models and information, provided system boundaries, level of detail, compatibility between systems, and compatibility with event trees for developing fault trees. Because of the consistency throughout the process and the linking and merging capabilities of the NUPRA code, dependencies, common cause effects, and system interaction were fully accounted for by the IPE. A more detailed description of the fault tree models is included in the WNP-2 IPE revision 1 submittal report (Reference 1.5.3).

NUPRA, a PC-based program developed by NUS Corporation, was used to quantify the seismic and internal fire accident sequences. It is an integrated program in that the databases, fault trees, event trees, quantification routines, sensitivity/uncertainty operations, and printing and plotting capabilities are all in one program. The fault tree linking approach is used to generate minimal cutset equations for various fault trees and accident sequences. NUPRA was verified and validated in accordance with the following NUS procedures:

CD-OP 5.1: "Computer Software Development Documentation, and Control"

CD-QAP 3.5: "Documentation, Verification, and Control of Software Programs"

It is the policy of the NUS Corporation to perform work related to nuclear power plant safety to the requirements of Title 10 of the Code of Federal Regulations, Part 50, Appendix B. After NUPRA was installed on in-house PCs, NUPRA fault tree and sequence results were checked against CAFTA and SETS results respectively. They were found to be identical. NUPRA has been verified and validated in accordance with the Supply System's computer code V&V procedure.

The seismic analysis utilized a second computer code, EQESRA, to combine component fragilities in boolean expressions to yield plant fragility, and to convolve the plant fragility with the seismic hazard curve to yield probability distribution on failure frequency.

EQESRA was verified and validated in accordance with the EQE Quality Assurance Manual dated 11/15/91, rev. 2.

EQESRA was also installed on Supply System in-house PCs. EQESRA boolean operations, uncertainty discretization, fragility curve condensation, and convolution capabilities were validated using known input problems. The output data obtained by the Supply System were found to be identical with the published output. EQESRA has also been verified and validated in accordance with the Supply System's computer code V&V procedure.

The remaining external events were evaluated using the progressive screening approach recommended in Generic Letter 88-20, supplement 4. The process involved reviewing plant specific hazard data and licensing bases documents, and identifying any significant changes that may have occurred since the operating license was issued. If the plant met the 1981 SRP criteria, no further evaluation was necessary. If the hazard frequency was acceptably low, no further analysis was necessary. Last, if a bounding analysis had been performed showing safe operation of the plant, no further evaluation was necessary. All of the external events other than internal fire and seismic activity were screened from further evaluation at one of the three screening stages in the progressive screening approach.

2.4 Information Assembly

Information gathering for the seismic IPEEE began with the development of the safe shutdown equipment list (SSEL) for seismic vulnerability evaluation. This list was developed starting with the equipment modeled in the WNP-2 IPE. The containment isolation valves and all safety related motor control centers, power panels, and instrument racks were added to this list. For each piece of equipment, location and description information was obtained from the plant Master Equipment List (MEL). This list includes all of the major pieces of equipment with a significant bearing on plant safety.

Because the IPE model used as the SSEL basis often was not resolved down to a level of detail necessary to pick up all of the support relays. The plant MEL was utilized to create a list of all of the chatter sensitive devices in 21 plant systems. These systems are as follows:

Condensate (COND)
Containment Instrument Air (CIA)
Containment Nitrogen (CN)
Control and Service Air (CAS)
Diesel Generator Building Environmental Control (DMA)
Electrical Distribution (E)
Fuel Pool Cooling (FPC)
Fire Protection (FP)
High Pressure Core Spray (HPCS)
Low Pressure Core Spray (LPCS)
Main Steam (MS)
Plant Service Water (TSW)
Reactor Closed Cooling (RCC)
Reactor Core Isolation Cooling (RCIC)
Reactor Feedwater (RFW)
Residual Heat Removal (RHR)
Reactor Protection System (RPS)
Reactor Building Environmental Control (RRA)
Standby Liquid Control (SLC)
Seal Steam (SS)
Standby Service Water (SW)

A total of 3550 chatter sensitive devices were added to the relay list. Location, seismic qualification references, and description information was also taken from the plant MEL. The list was screened first for unnecessary functions based on the component description. For those devices remaining, information was gathered from the Qualification Information Document (QID) pertaining to the device.

The new hazard analysis and soil structure interaction analysis was performed by Geomatrix Consultants. These analyses utilized soil composition information taken from the WNP-2 FSAR and regional seismic history data.

Three walkdowns were performed in support of the seismic fragility analysis. These walkdowns were documented using the forms developed for the Seismic Margins Assessment as presented in EPRI NP-6041-SL, rev. 1 (Reference 2.5.1). Each of the walkdowns were conducted by at least two trained seismic experts (provided by EQE International) and at least one PSA analyst from the Supply System. Walkdowns involving building structures and mechanical equipment also included a civil/structural expert from the Supply System. Additional information on individual components was taken from the QID file pertaining to the component and/or plant structural drawings.

The SSEL data base for the fire IPEEE was compiled using three equipment lists as sources; the appendix R safe shutdown equipment list, the WNP-2 IPE basic event equipment, and the valves required for containment isolation. For Appendix R equipment, cable routing and cable protection data were available in the Appendix R analysis. For the remaining components, the Cable and Raceway Penetration Schedule (CARPS) was utilized to obtain cable routing information.

The SSEL for the seismic IPEEE and the SSEL for the fire IPEEE differ. The origin of both lists is the same, the equipment which was determined to have an impact on core damage in the IPE. Both lists also include the valves required to isolate primary containment. The lists diverge from that point because of the differences in which fire and earthquake affect equipment, and the methods available to perform the analysis. In the fire analysis, electrical cable and routing carry primary importance. Thus, required electrical cable from each major piece of equipment was traced to its power source. This process automatically picked up the location of supporting subcomponents, such as relays and switches without having to call them out explicitly on the list. Seismic events, on the other hand, affect structures. So, structures like cabinets and racks were included in the seismic SSEL to assure that any important subcomponents within would be accounted for if the structure was found to fail the seismic screening criteria.

Ignition source and combustible loading and fire protection system information was gathered primarily from two sources. Information was compiled from the Fire Protection Evaluation presented in Appendix F of the WNP-2 FSAR (Reference 2.5.2). In addition, a walkdown of each of the fire areas was performed documenting the number and location of ignition sources, location of combustibles, and confirming fire protection features. Information compiled during the walkdown are part of the second level of documentation retained at the Supply System.

An additional walkdown was performed to assess the impact of fire/seismic interactions. This walkdown was a joint effort utilizing a seismic expert from EQE International, a fire protection expert from VECTRA and a PSA analyst from the Supply System. The results of this walkdown are presented in section 4 of this report and the walkdown report is part of the Tier 2 documentation retained at the Supply System.

Information for the remaining external events was gathered from published reports and/or correspondence between the Supply System and the NRC. In addition, information was gathered from local meteorological stations and transportation facilities to update and/or confirm information in the Supply System licensing documents. A walkdown to confirm essential details in support of the other external events evaluation was performed by the Supply System.

2.5 References

2.5.1 EPRI, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," EPRI NP-6041-SL, Project 2722-23, August 1991

2.5.2 Supply System, "WPPSS Nuclear Project No. 2 Final Safety Analysis Report," Appendix F

3.0 SEISMIC ANALYSIS

This section summarizes the results of the seismic analysis performed in response to the guidelines contained in Generic Letter 88-20 (Supplement 4) and NUREG-1407. Because the seismic analysis involves a significant amount of detail, only information necessary to summarize the results is included in this main report. The complete documentation on the detailed analyses, including calculations, detailed walkdown results, soil-structure interaction assessment, hazard analysis, and fragility development, is included in the Supply System's files. The detailed analyses are referenced as appropriate within this section.

The WNP-2 seismic IPEEE was completed in accordance with the guidelines provided in NUREG-1407 and NUREG/CR-2300. For the seismic evaluation, the Washington Public Power Supply System selected the Seismic Probabilistic Safety Assessment (SPSA) approach.

The SPSA for WNP-2 was primarily based on the methodology and models used in the WNP-2 Individual Plant Examination for Internal Events (IPE for Internal Events). The internal events IPE uses a fault tree linking approach which allows explicit modeling of system/component dependencies. This feature provides a direct capability to propagate seismic failures through the front-line mitigation systems. The SPSA also evaluates the performance of the containment, explicitly investigating the potential for containment release scenarios which are unique when compared to those evaluated for the internal events IPE.

Major inputs to the SPSA included the results and insights obtained from the plant walkdowns conducted in support of the IPEEE. The walkdowns were conducted in accordance with the guidance included in the Generic Implementation Procedure (Reference 3.6.1) and the EPRI Seismic Margins Methodology (Reference 3.6.2). A relay chatter evaluation was also conducted in accordance with NUREG-1407. The results of the relay chatter evaluation were factored into the SPSA model.

A plant-specific hazard curve was developed and utilized for the WNP-2 site. A summary of the hazard evaluation is contained in Section 3.1.1.

Overview of Methodology

The key elements of a seismic PSA are:

1. Seismic hazard analysis - estimation of the frequency of various levels of seismic ground motion (acceleration) occurring at the site;

2. Seismic fragility evaluation - estimation of the conditional probabilities of structural or equipment failure for given levels of ground acceleration;

3. Systems/accident sequence analysis - modeling of the various combinations of structural and equipment failures that could initiate and propagate a seismic core damage accident sequence;

4. Evaluation of core damage frequency and public risk - assembly of the results of the seismic hazard, fragility and systems analyses to estimate the frequencies of core damage and plant damage states; assessment of the impact of seismic events on the containment integrity; and integration of these results with the core damage analysis to obtain estimates of seismic risk in terms of effects on public health.

Each of these steps is discussed in more detail in the following sections.

Summary of Results

The plant was conservatively designed for a 0.25g peak ground acceleration and for the most part has very large margins such that most items can be screened out from further evaluation (i.e., do not require detailed fragility evaluations). Detailed walkdowns conducted jointly by plant engineering staff and industry experts were performed on three separate occasions. A few items were noted in the walkdowns and in the subsequent review of seismic qualification documentation which appeared to be governing. For these items, detailed fragility evaluations were performed. Identified spatial systems interactions included a few cases of possible cabinet impacts which could have affected relay performance. No spray or flood issues were noted although there is one case where failure of a liquid tank could possibly flood the diesel generator air inlets with gaseous N_2. The nitrogen tank failure possibility had previously been evaluated by deterministic analysis and shown to have a maximum possible impact of 8 minutes on the emergency diesels. There were no significant seismic/fire interactions noted.

New probabilistic soil-structure interaction analyses were performed which demonstrated a very large margin in the original Safe Shutdown Earthquake (SSE) in-structure spectra and structural loads. Based upon a comparison of structural loads and some supporting structural calculations, all structures were screened out. No unique containment vulnerabilities were identified.

A soil liquefaction analysis and a seismic soil settlement analysis were performed (Reference 3.6.14). The soil liquefaction analysis was performed using the guidelines contained in EPRI NP-6041. It was concluded that the liquefaction potential for WNP-2 soils is negligible, even if they were to become saturated. The seismic soil settlement analysis indicated that seismic incurred settlements of WNP-2 structures are negligible.

A comparison of SSE and new probabilistic spectra, comparisons of qualification margin relative to the SSE and a detailed plant walkdown enabled most equipment to be screened out. The exceptions to this were certain motor control centers and a few Instrument and Control (I&C) cabinets in other locations with high seismic demand. The motor control centers are the controlling case at the locations of highest seismic demand having a High Confidence of Low Probability of Failure (HCLPF) of 0.43g for function during the earthquakes. These same configurations in other lower seismic demand locations were screened out.

The overall impression from the walkdowns and the review of seismic qualification documentation is that the plant is very well constructed and has a high resistance to seismic loading. The seismic qualification documentation demonstrated a high degree of conservatism in almost all cases. The few cases where this high degree of conservatism did not exist were noted for detailed fragility assessments for the SPSA.

The results of the SPSA indicate that seismic induced loss of offsite power (LOOP) sequences are the dominant contributors to seismic plant fragility. LOOP sequences contribute about 77% of the overall seismic core damage frequency of 2.0×10^{-5}/yr. Seismic induced loss of critical switchgear cooling accounts for another 18% of the total seismic CDF. The calculated seismic CDF is a conservative estimate and no recovery actions have been modeled.

The SPSA effort did not identify any major weaknesses in the design, construction or maintenance practices at WNP-2. During the seismic team walkdowns a handful of interaction and anchorage detail issues were raised. None were safety significant, and all have either been corrected or are being evaluated for correction.

3.1 Hazard Analysis

Probabilistic Seismic Hazard Analysis (PSHA) is defined as an evaluation of the probability or likelihood that various levels of ground motion will be exceeded during a specified time period. Seismic hazard is usually expressed in terms of the frequency distribution of the peak value of a ground motion parameter (e.g., peak ground acceleration) during a specified time interval. The different steps of this analysis are as follows:

1. Identification of the sources of earthquakes, such as faults and seismotectonic provinces;

2. Evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities;

3. Development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., peak ground acceleration) at the site;

4. Integration of the above information to estimate the frequency of exceedance for selected ground motion parameters.

The hazard estimate depends on uncertain estimates of attenuation, upperbound magnitudes, and the geometry of the postulated sources. Such uncertainties are included in the hazard analysis by assigning probabilities to alternative hypotheses about these parameters. A probability distribution for the frequency of occurrence is thereby developed. The annual frequencies for exceeding specified values of the ground motion parameter are displayed as a family of curves with different probabilities.

A Bayesian estimate of the frequency of exceedance at any peak ground acceleration was obtained. Thus, the weighted sum of the frequencies of exceedance for a given acceleration are determined by the different hazard curves; the weighting factor is the probability assigned to each hazard curve.

For the WNP-2 SPSA, a site-specific hazard estimate was developed. The hazard estimate development process and results are included in the hazard report which is held as a Supply System permanent record. (Reference 3.6.3)

The probability that at a given site a ground motion parameter, Z, will exceed a specified level, z, during a specified time period, t, is given by the expression:

$$P(Z>z|t) = 1.0\text{-}e^{-v(z)*t} \leq v(z)*t$$

where v(z) is the average frequency during the time period t at which the level of ground motion parameter Z resulting from earthquakes on all sources in the region exceeds z at the site (i.e., v(z) is the frequency of exceedance). The inequality at the right of the equation is valid regardless of the appropriate probability model for earthquake occurrence, and v(z)*t provides an accurate and slightly conservative estimate of the hazard for probabilities of 0.1 or less, provided v(z)*t is the appropriate value for the time period of interest.

The frequency of exceedance, v(z), is a function of the randomness in the time, size, and location of future earthquakes and the randomness in the level of ground motions they may produce at the site. The frequency is computed by combining an expression which represents the frequency of earthquakes on the likely sources, with expressions representing the probability density functions for event size, distance to the earthquake rupture, and likelihood that the ground motion exceeds level z given the distance to, and magnitude of the earthquake. Reference 3.6.3 provides details on the development of the expressions.

The probability functions contained within the expressions used to model the frequency of exceedance (v(z)) represent the uncertainties inherent in the natural phenomena of earthquake generation and seismic wave propagation. For the WNP-2 region (and usually for any region) there are considerable uncertainties which arise from limited data and/or alternative interpretations of the available data. This site-specific study explicitly incorporates these additional uncertainties into the analysis to provide a quantitative assessment of the uncertainty in the seismic hazard estimate.

The uncertainty in modeling the natural phenomena is incorporated into the hazard analysis through the use of a logic tree methodology employed in previous studies.[1] The logic tree formulation for seismic hazard analysis involves specifying discrete alternatives for states of nature (analysis model parameter values i.e., attenuation relationships, source spatial distribution, and frequency data) and judgments on the relative likelihood that each discrete alternative is correct. The relative likelihoods of the different parameter values are typically based on subjective judgment, but may be specified by an objective statistical analysis if available data warrant an assessment. The logic trees developed for this study are summarized in the next section.

3.1.1 Seismic Hazard Model

The seismic hazard at a site is as function of the location and geometry of potential sources of future earthquakes, the frequency of occurrence of various size earthquakes on these sources, and the characteristics of seismic wave propagation in the region. For this study, these elements are analyzed within a framework that addresses both the randomness of the earthquake process and the uncertainty in modeling the process. The model consists of two basic components: a model of the sources of potential future earthquakes, and a model of the effects at the site due to the future earthquakes.

Figure 3.1-1 displays the overall logic tree representing the seismic hazard model developed for this study. The logic tree is laid out to provide a logical progression from general aspects/hypotheses regarding the characteristics of seismicity and seismic wave propagation in the region to specific input parameters for individual faults and fault segments. The tree is discussed in detail in Reference 3.6.3; it is summarized briefly here.

The first node of the tree represents uncertainty in selecting the appropriate strong ground motion attenuation relationship. The tree then expands into subtrees, one for each of the seismic source types included in the analysis for the WNP-2 site. To the right of this node, each source is considered to be acting independently. The distribution in the total computed hazard is obtained by convolving the independent distributions obtained for each seismic source.

The following nodes include: source activity, source tectonic model, segmentation, source geometry, maximum magnitude, earthquake recurrence, and magnitude distribution models. The specific node levels used and the resulting branches and associated relative likelihoods are dependent on the type of source assessed (see Reference 3.6.3).

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[1] Power, M.S., et. al., "Seismic Exposure Analysis for the WNP-2 and WNP-1/4 Site: Appendix 2.5K to Amendment N.18, WNP-2 FSAR," September 1981; Woodward-Clyde Consultants, "Seismic Hazard Assessment for the Hanford DOE Site: Report prepared for Westinghouse Hanford Company," WHC-MR-0023, 1989.

3.1.2 Attenuation Relationships

Strong ground motions produced by earthquakes are influenced by the characteristics of the earthquake source, the crustal wave propagation path, and the local site geology. At present, no strong motion data have been recorded in the study region with which one can evaluate these characteristics. Therefore, empirical attenuation models from regions considered to have similar characteristics have been used to evaluate the ground motion hazard.

The WNP-2 site lies at the eastern edge of the plate boundary between North America and the Pacific and Juan de Fuca plates. The limited data from earthquakes occurring to the west of the site (in northern Oregon) indicate that the source characteristics are similar to those of California earthquakes.[2] On the basis of the studies, it was judged that the empirical strong motion models based primarily on California strong motion data would be appropriate to represent the effects of source and travel path for eastern Washington earthquakes.

Three sets of attenuation relationships were selected for use in characterizing the ground motions at the WNP-2 site:

1. Yoyner, W.B., and D.M. Boore, "Prediction of Earthquake Response Spectra," U.S. Geological Survey Open File Report 82-977, 1982;

2. Sadigh, K., et. al., "Specification of Ground Motion for Seismic Design of Long Period Structures," abstract, Earthquake Notes, v. 57, n. 1, p. 13. 1986;

3. Campbell, K.W., "Empirical Prediction of Near-Source Ground Motion from Large Earthquakes," Proceedings of the International Workshop on Earthquake Hazard and Large Dams in the Himalaya, New Delhi, January 15-16, 1993.

The WNP-2 IPEEE study also used the most up-to-date attenuation relationship developed for estimating ground motions on firm soil sites from subduction zone earthquakes, namely:

> Crouse, C.B., "Ground-Motion Attenuation Equations for Earthquakes on the Cascadia Subduction Zone," Earthquake Spectra, v. 7, p. 210-236, 1991.

This relationship is based on regression analysis of strong motion data recorded in Mexico, South America, Alaska, and Japan.

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[2] See for example University of Washington, "Annual Technical Report, 1986, On Earthquake Monitoring of Eastern Washington," Geophysics Program, U of W, Prepared under contract with USDOE, contract EY-76-S-06-2225, 1986.

3.1.3 Seismic Source Characterization

The seismic source characteristics are presented and incorporated into the seismic hazard analysis through the use of the logic trees introduced in Section 3.1.1. The WNP-2 site is subject to the activities and characteristics of two major influences: (1) the Yakima fold belt, and (2) the Cascadia subduction zone lying along the coast of Washington and Oregon.

The regional seismicity of the Yakima fold belt has been addressed through the compilation of an earthquake catalog. The catalog uses data which is presented in the WNP-2 FSAR for the period of 1850 through 1969, and from the University of Washington seismic records for the period 1969 through March 1991. The catalog contains a variety of magnitude measures, e.g., local magnitude (M_L), surface wave magnitude (M_S), moment magnitude (M), and coda-duration magnitude (M_C).

Earthquake activity in the Columbia Basin, central Washington, is attributed to three separate source regions of the siesmogenic crust: (1) fault sources expressed at the surface as the Yakima Folds and related thrust/reverse faults; (2) a shallow basalt source not spatially associated with the Yakima Folds, but which accounts for the observed seismicity within the Columbia River Basalt Group (CRBG); and (3) a crystalline basement source region. These three sources are assumed to account for all of the observed seismicity within the region. The seismic hazard source model for the Yakima Fold belt is developed to incorporate the data and characteristics of these three source regions. Reference 3.6.3 provides the analysis details.

The Cascadia subduction zone (an interface between two tectonic plates) lies along the west coast of North America from Cape Mendocino in northern California to mid-Vancouver Island. Subduction zones contribute additional independent seismic sources from the standpoint of source locations, maximum magnitudes, and earthquake recurrence. Therefore, for the WNP-2 IPEEE study, the Cascadia subduction zone influences are conservatively included in the assessment.

The plate interface lies at a distance of approximately 350 km from the WNP-2 site. Estimates for the frequency of occurrence of the largest magnitude events, based upon plate convergence rates and paleoseismic data, center around 500 years. Additional data is included in Reference 3.6.3.

3.1.4 Seismic Hazard Analysis Results

Seismic hazard calculations were made for peak horizontal ground acceleration and 5% damped response spectral accelerations at periods of 0.3 and 2.0 seconds. The computed mean peak hazard and the 5^{th} - 95^{th} percentile hazard curves for the site, for peak acceleration and 5% damped spectral acceleration at the 0.3 and 2.0 second periods, are shown in Figure 3.1-2. The uncertainty band varies from about one order of magnitude at low ground motion levels to over two orders of magnitude at large ground motion levels. The uncertainty in the computed hazard also increases as one considers longer periods of vibration. The distribution in computed frequency of exceedance becomes skewed at the higher ground motion levels, and the mean hazard lies near the 75^{th} percentile of the hazard distribution.

Based on the mean hazard curves, the following ground motion levels were computed for various return periods:

Peak Ground Acceleration (g) or 5% damped Spectral Acceleration (g) for Return Periods (yrs.) of:

Period	100	500	1,000	2,500	5,000	10,000	100,000
PGA	0.036	0.081	0.117	0.178	0.236	0.308	0.615
0.3 seconds	0.081	0.214	0.304	0.455	0.602	0.781	1.624
2.0 seconds	0.012	0.051	0.080	0.116	0.169	0.214	0.452

For WNP-2, the basement source is the largest contributor to the ground motion (as compared to the folds and shallow basalt). The distant Cascadia interface source is the dominant contributor to long period (2 second spectral acceleration) ground motion hazard.

Figure 3.1-3 shows the relative contribution of events in different magnitude intervals to the computed mean hazard. The plot presents a histogram of the percent contributions of events in 0.25 magnitude unit-wide intervals. Histograms are presented for peak acceleration and spectral acceleration at periods of 0.3 and 2.0 seconds for mean annual frequencies of exceedance of 10^{-3} and 10^{-4} (return periods of 1,000 and 10,000 years).

The distributions in the computed hazard shown in Figure 3.1-2 represent the cumulative effect of all levels of parameter uncertainty included in the hazard model logic tree (Figure 3.1-1). Reference 3.6.3 includes a discussion of the relative contribution of various components of the hazard model logic tree to the overall uncertainty.

The seismic hazard computed in Reference 3.6.3 and used for the WNP-2 IPEEE is somewhat higher than that computed from previous models (e.g., Woodward-Clyde Consultants, "Seismic Hazard Assessment for Hanford DOE Site," WHC-MR-0023, 1989). The reasons for these differences are: (1) use of multiple attenuation models, some with higher levels of dispersion; (2) updated estimates of earthquake occurrence; and (3) inclusion of additional sources of potential earthquakes (the basement source and the Cascadia interface).

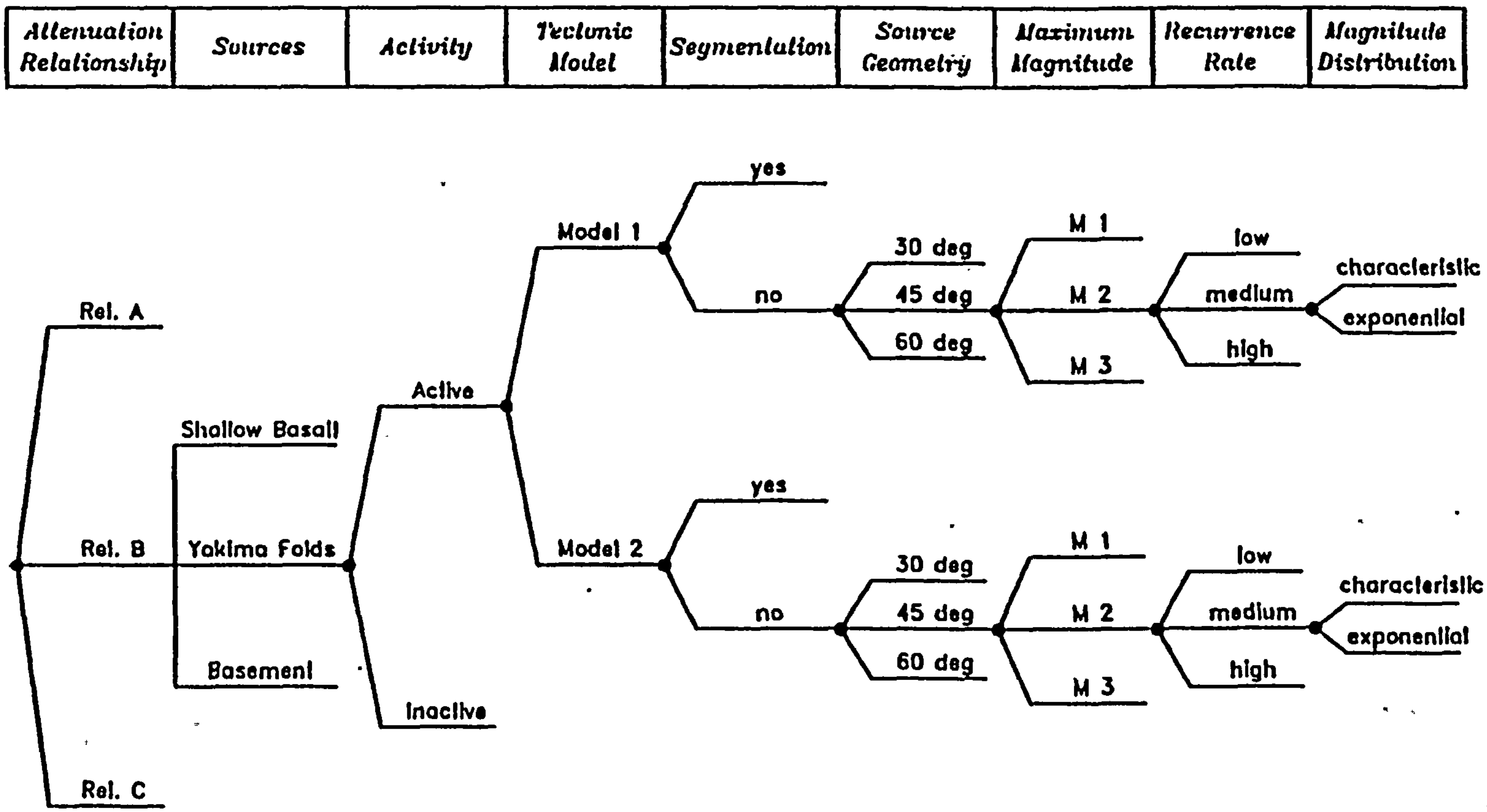

Figure 3.1-1
Seismic Hazard Model Development

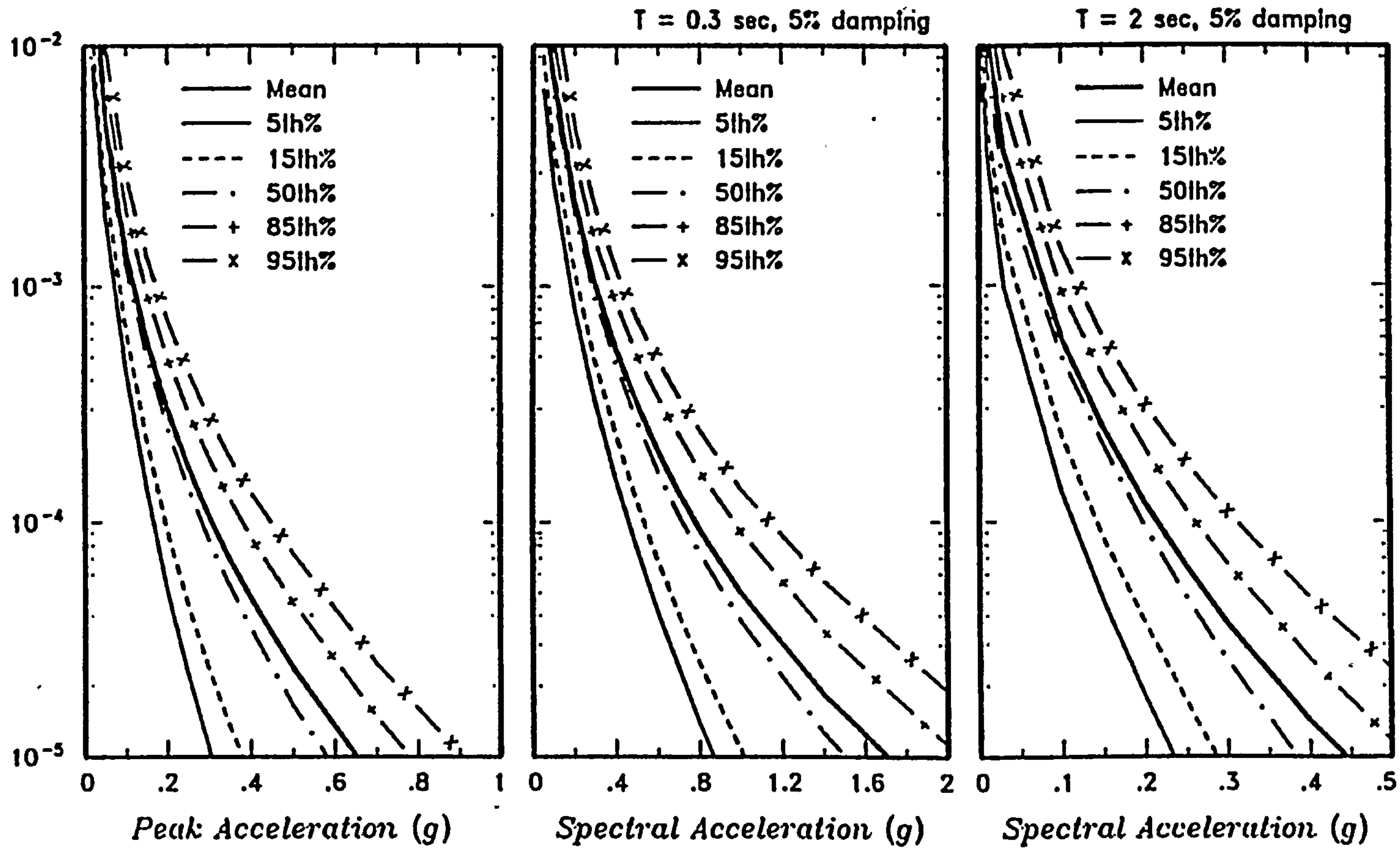

Figure 3.1-2
Cumulative Seismic Hazard Curve for WNP-2

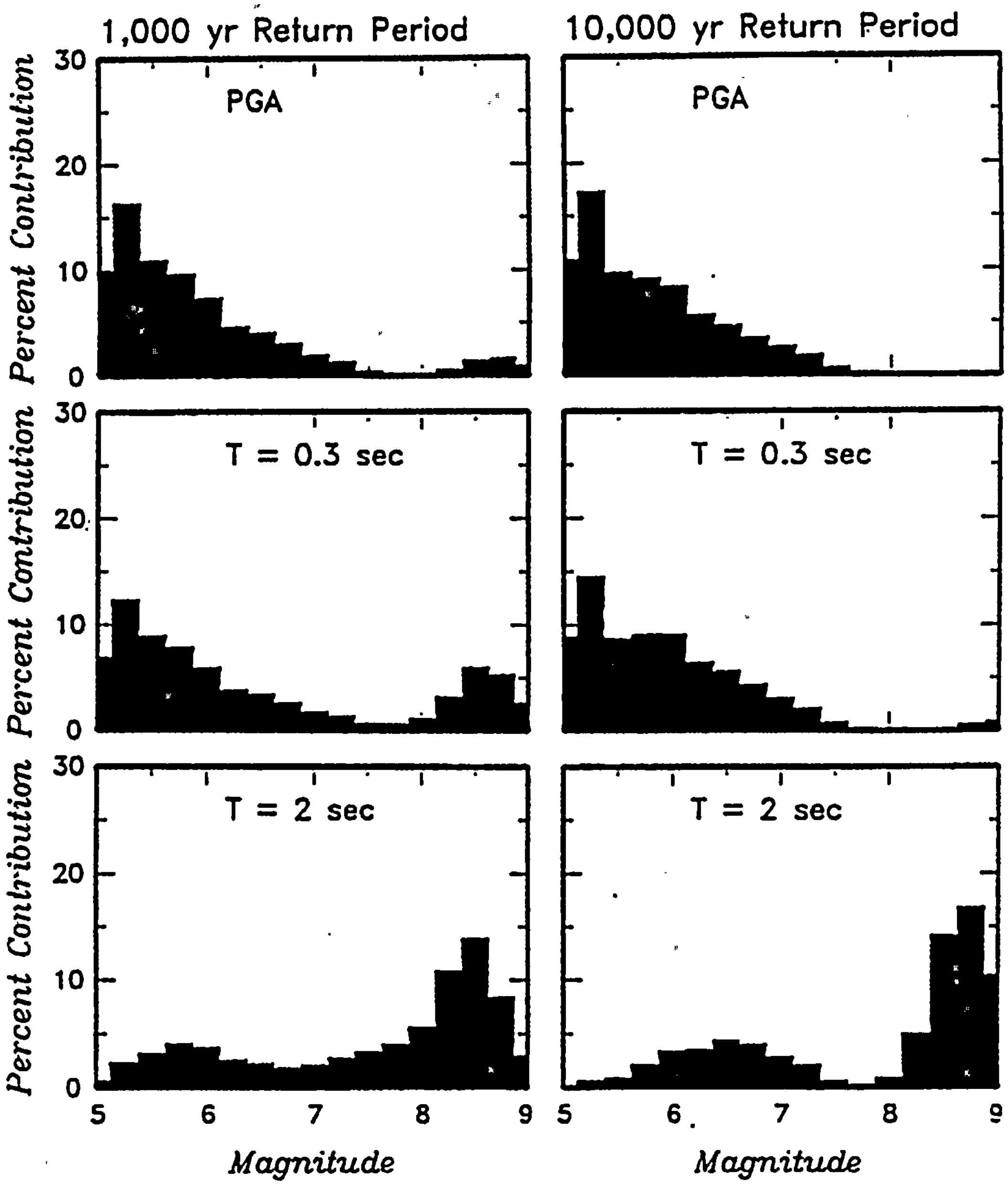

Figure 3.1-3

Relative contribution of events in various magnitude intervals to mean hazard. Shown are results for peak horizontal acceleration and 5% damped spectral acceleration at 0.3 and 2.0 seconds.

3.2 Review of Plant Information and Plant Walkdown

In the IPEEE, an evaluation is performed for a new earthquake which is generally larger than the Safe Shutdown Earthquake (SSE). However, the evaluation methods to determine the seismic margin inherent in the design tend to be more optimistic and realistic such that the original design is generally shown to have a great deal of margin above the SSE. This is the case for WNP-2. To develop an understanding of the sources of conservatism in the original design, comparisons are made of the original design basis and the methodology used in the development of fragilities for the seismic PRA.

3.2.1 Earthquake Input Motion

WNP-2 was designed for a SSE of 0.25g peak ground acceleration. The ground motion spectral shape was similar to a Regulatory Guide 1.60 standard spectral shape but did not contain the degree of amplification that is inherent in a RG 1.60 spectral shape. The amplification is closer to a median amplification than for a plus one standard deviation amplification which is the basis for the RG 1.60 spectrum. The horizontal SSE spectrum is shown in Figure 3.2-1. The vertical spectrum was specified as 2/3 of the horizontal spectrum. The Operating Basis Earthquake (OBE) was specified as 1/2 of the SSE.

For the IPEEE, Uniform Hazard Spectra (UHS) were developed (Section 3.1). At each earthquake level, with an associated probability of occurrence, a ground motion spectrum is developed for which every point on the spectrum has an equal probability of occurrence. Figure 3.2-2 shows a 10,000 year return period UHS. The amplification of the peak ground acceleration of the UHS is slightly greater than the SSE spectrum.

3.2.2 Damping

The damping values used for WNP-2 design are generally more conservative than those listed in USNRC Regulatory Guide 1.61. For piping, the RG 1.61 damping values could be used if the FSAR values resulted in difficulty in qualification. For IPEEE purposes, the damping associated with the appropriate response level is used to calculate median capacities. If the failure mode is functional without inelastic deformation, lower damping values are used than if the failure mode is purely structural. The guidance of EPRI's seismic fragility methodology (Reference 3.6.4) was used for damping determination.

Table 3.2-1 compares design damping with damping levels recommended by EPRI for calculation of median fragilities. There is of course variability in damping and this variability is taken into account in the development of fragilities and probabilistic structural response. Where values are not recommended in Reference 3.6.4, the values assumed for the fragility analysis and screening are listed in Table 3.2-1.

3.2.3 Capacities

The design of WNP-2 followed industry codes and standards as adopted by the USNRC. Structures were designed to nominally remain elastic. Equipment and piping designed to ASME codes may exceed elastic limits but demonstration of functionality was also a requirement for active equipment and piping. Material properties used in design are code specified values that are set at about the 95% confidence level. Equipment qualified by test theoretically has the least margin. The tested equipment are subjected to Test Response Spectra (TRS) that must envelop the Required Response Spectra (RRS) defined by the amplified floor response spectra at the equipment mounting location. There is no required safety factor on the test input and no knowledge obtained from the tests as to the capacity beyond the test level, thus, the conservatism in the floor spectra generation is the only definable margin although significant margin may exist beyond the achieved test level. In some cases results of fragility tests, typically of relays, are used. The fragility level, which may be defined as the threshold of malfunction or merely the manufacturers highest achieved test level, is compared to the demand but no factor of safety is required.

The net result of the design and qualification process is that there are inconsistent margins above the qualification level. Ductile structural failure modes would tend to have the larger margins whereas non-ductile failure modes or equipment function demonstrated by testing would tend to have the smallest demonstrated margin. Therefore, because the real capacity is not determined by a proof test, the actual margin from testing is not quantifiable.

In developing fragilities, the knowledge of the conservatisms in the design process is utilized to determine the median capacity. Median material properties are utilized rather than code properties. For ductile structural failures, the concept of ductility is utilized to define the margin beyond the elastic limits of the structure or equipment. The median capacity is well beyond the elastic limit and applicable ductility factors are derived to reflect this. For ductile modes of failure, the ductility factor of safety is discussed in Appendix A of Reference 3.6.5.

For non-ductile structural failure modes, the margin is defined as the median ultimate strength divided by the applied load or stress. Non-ductile failure modes tend to have less margin than ductile modes if the loading is at code limits, but there are usually large margins between the design loads and allowable loads such that the real margin is significantly more than the design codes would imply.

The margin for tested components is developed from guidance in Reference 3.6.4. This guidance results from a consensus of several experts in equipment qualification testing. A single proof test does not demonstrate a High-Confidence-of-Low-Probability-of-Failure (HCLPF) which is an industry measure of acceptance for demonstrating adequate seismic margin. Thus, without a significant margin in the RRS or a significant overtest, equipment qualified by test would be computed to have low margin above the test level. The industry practice is to significantly overtest at frequencies beyond the frequency of the peak of the RRS. An example is shown in Figure 3.2-3. Since WNP-2 is a soil site, the peaks of the RRS occurred at low frequencies and most equipment was significantly overtested. This overtest is incorporated into the fragility development.

3.2.4 Walkdown

The walkdown is emphasized in NUREG-1407 as being a particularly important ingredient to the IPEEE seismic evaluation and is recommended to be conducted per the guidance of Reference 3.6.2. The Supply System utilized the services of a seismic consulting company, EQE, International, to assist with the walkdowns conducted for the WNP-2 IPEEE. EQE and the Supply System performed three walkdowns of WNP-2 to ultimately examine in detail at least one of the dual train components internally and all components externally. Documentation of the walkdowns was done using the forms recommended by EPRI and is retained in Supply System permanent records (Reference 3.6.7).

Walkdowns are performed for the purpose of identifying equipment/system seismic weak links in either the component load path or anchorage, potential seismic failure of non-safety items resulting in falling and proximity interactions, potential flooding or fluid spray interactions and potential seismic fire interactions. For a PRA, seismic fragilities must be developed for those items which, based upon the walkdown and reviews of the design basis and equipment qualification, appear to be the weaker links. These fragilities are then utilized, along with the seismic hazard and plant function models to compute the probability of core damage.

To support the WNP-2 IPEEE, and per the guidelines included in NUREG-1407, the Supply System developed a Safe Shutdown Equipment List (SSEL). The SSEL includes the Systems, Structures, and Components (SSCs) necessary to achieve safe shutdown. For seismic events, the SSEL must contain the SSCs necessary to respond to certain postulated events, such as:

- Mitigate the effects of a Loss Of Offsite Power (LOOP),
- Mitigate the effects of a Small Break Loss Of Coolant Accident (SBLOCA), and
- Ensure containment performance functions and the ability to isolate containment.

The SSEL used as its foundation the SSCs included within the internal events IPE. Additional SSCs were added to the SSEL to take into consideration the potential seismically-induced failures of passive components whose failures were not considered to be probabilistically significant during the quantification of the internal events IPE. An iterative process was employed to complete the SSEL: PRA analysts, seismic experts, and walkdown participants continually reviewed the items on the SSEL to ensure that it was complete. Section 6 of Reference 3.6.5 contains the SSEL developed for the WNP-2 SSEL, along with the derived fragility or the screening disposition of each item on the SSEL.

The initial walkdown of the inside of containment was performed May 26 through June 3, 1993 during the refueling outage. After completion of the containment walkdown, a walkby of the essential equipment outside of containment was conducted to familiarize the team with the overall plant layout and to develop a feel for the relative ruggedness of the plant equipment. At the time of the first walkdown the SSEL for systems outside of containment was not complete. During the containment walkdown, all of the mechanical penetrations were examined to determine if there were any potential issues with containment performance associated with differential motions between the containment and the internal concrete structure (reactor pedestal and sacrificial shield wall), due to spatial systems interactions or other concerns such as with piping supports. All equipment and distributive systems inside containment appeared robust and no concerns were noted. The main steam isolation valves were noted to be well outside of the valve seismic experience database and it was noted to examine the seismic qualification before it was concluded that the valves could be screened out.

A second walkdown was performed during plant operation from September 21 through 24, 1993. The SSEL at this point was well developed, but was still not complete. The mechanical and electrical power equipment were fairly firmed up but the I&C equipment was not finalized. Because of the operating status of the plant, many electrical panels could not be accessed for a detailed examination. The walkdown focussed on gathering data and identifying components for which the qualification documentation was requested for further review. Many components were screened out on the basis that their ruggedness appeared to be much greater than other components identified for further review. Between this walkdown and the final walkdown from May 6 to May 13, 1994, many seismic qualification documents were retrieved and reviewed to gain more insight into the seismic ruggedness of equipment prior to the final walkdown.

During the final walkdown in May 1994, one train of electrical equipment was deenergized so that access was available to the inside of at least one of identical dual train components. However, HPCS and RCIC are single train systems. Consequently RCIC was maintained on line for shutdown cooling backup so the electrical and I&C panels associated with RCIC were generally restricted.

The SSEL was completed prior to the final walkdown. During the walkdown some minor modifications were made to delete equipment which had no contribution to safe shutdown.

Based upon the walkdown observations and prior and subsequent review of a significant amount of seismic qualification data, most equipment items could be screened out. There were only a few items noted in the walkdowns that were a concern and warranted fragility development. The motor control center anchorage configuration was the most notable. This is discussed in detail in References 3.6.5, 3.6.9, and 3.6.10 where the fragility derivation is described. A few instances were noted of potential cabinet impact which could only affect the performance of relays. These cases are noted in the SSEL and fragility tabulation in Reference 3.6.5.

Part of the IPEEE assessment is to address internal fires. This was performed by the Supply System with assistance from the VECTRA Corporation (formerly ABB Impell Corporation). The potential for seismic/fire interaction is included in this evaluation. A joint walkdown of the fire suppression systems was performed by EQE, VECTRA and WPPSS personnel on September 28 and 29, 1993. The potential for seismic induced fire initiation, seismic induced actuation of fire suppression systems and seismic induced degradation of fire suppression systems was examined. EQE prepared a report which was incorporated into the Fire IPEEE report prepared by VECTRA (see Section 4.0 for discussion of the Fire IPEEE). It was concluded from the seismic/fire interaction walkdown that:

- No sources of seismic induced fire were noted.

- For inadvertent actuation it was noted that, in areas of safety related equipment, the fire protection systems are preaction and require redundant sensor/relay circuit closure to result in actuation of deluge valves and local head actuation for water release.

- Some vulnerabilities exist in the availability of the suppression systems. Batteries for the diesel driven fire pumps were not well supported and common pumps and headers are used to service both safety and non-safety areas.

The seismic/fire interaction walkdown data are contained in the documentation prepared subsequent to the walkdowns and are retained in Supply System permanent records. (Reference 3.6.6)

Table 3.2-1

Comparison of Design Damping with Median Damping Recommended for Fragility Development

Structure or Component	Safe Shutdown Earthquake	Med Damping for Fragilities
Welded Steel Plate Assemblies	1.0	**
Welded Steel Frame Structures	3.0	7
Bolted or Riveted Steel Frame Structures	3.0	7
Reinforced Concrete Equipment Supports*	3.0	3-5
Reinforced Concrete Structures*	5.0	3-10
Vital Piping - ****	1.0	5
Equipment	2.0	3-5***
Welded Structural Assemblies (Equipment and Supports)	2.0	5
Reactor Pressure Vessel, Support Skirt, Shroud Head, Separator and Guide Tubes	2.0	**
Control Rod Drive Housings	3.5	**
Fuel	7.0	**

* Range depends upon level of response, degree of concrete cracking.

** Not addressed in Reference 3.6.4; 5% would be used for equipment supports and reactor internals.

*** For electrical cabinets containing relays, 3% damping is best estimate if cabinet response is linear elastic.

**** RG 1.61 values could be used for design if the value listed was too restrictive.

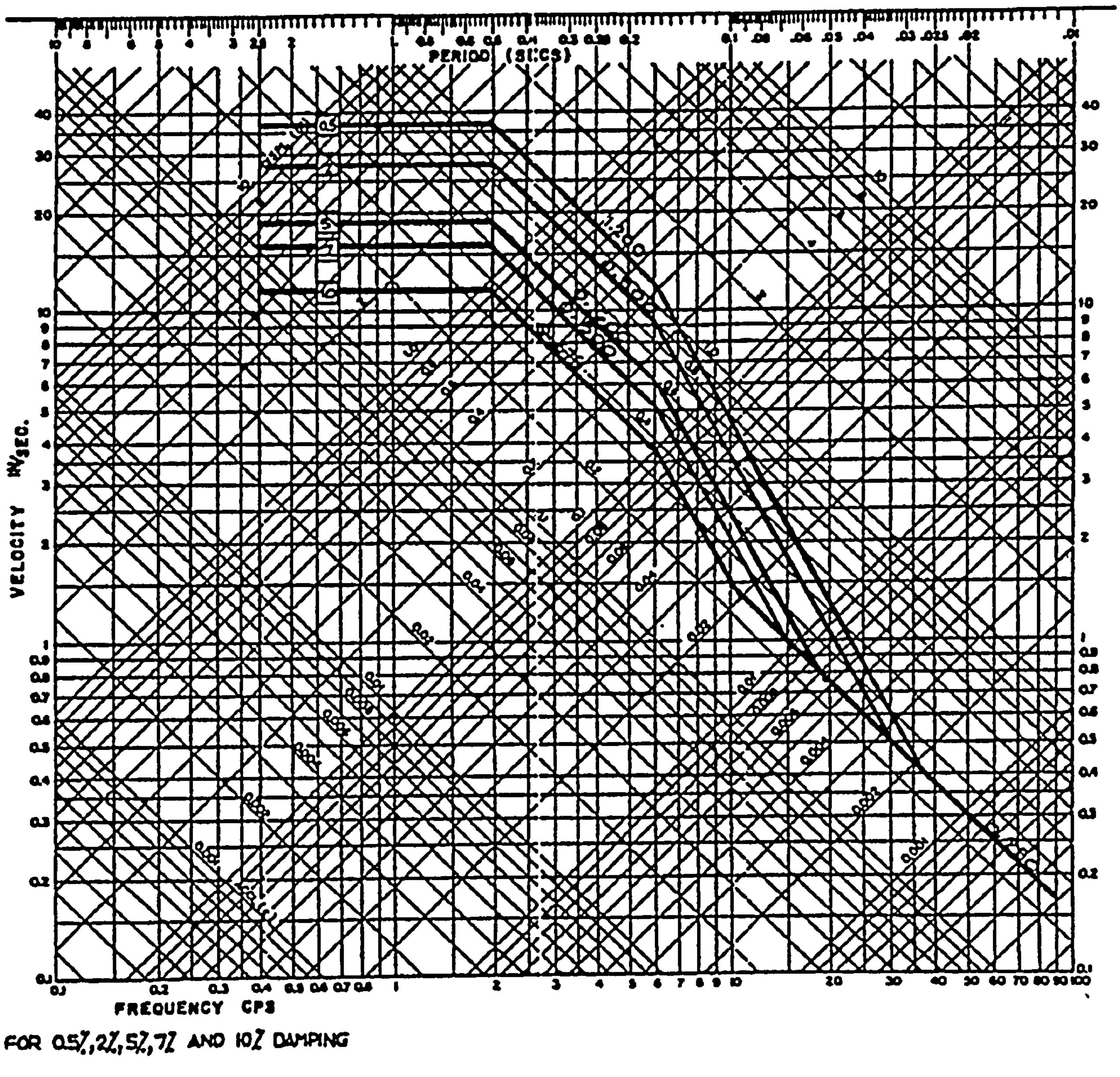

Figure 3.2-1
Response Spectra Safe Shutdown Earthquake Horizontal Component

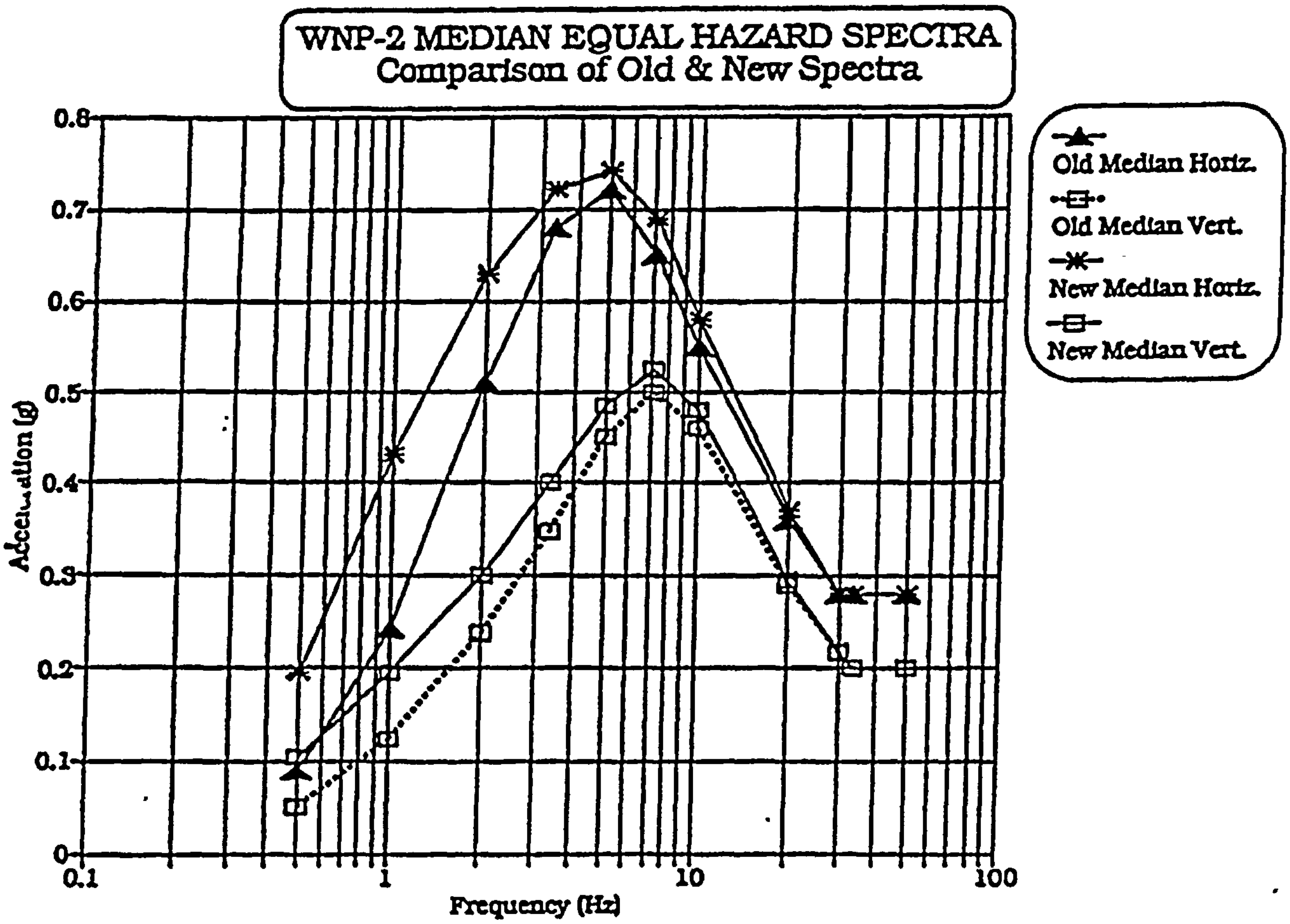

Figure 3.2-2
WNP-2 10,000 Year Return Period Uniform
Hazard Spectra, 5% Damping

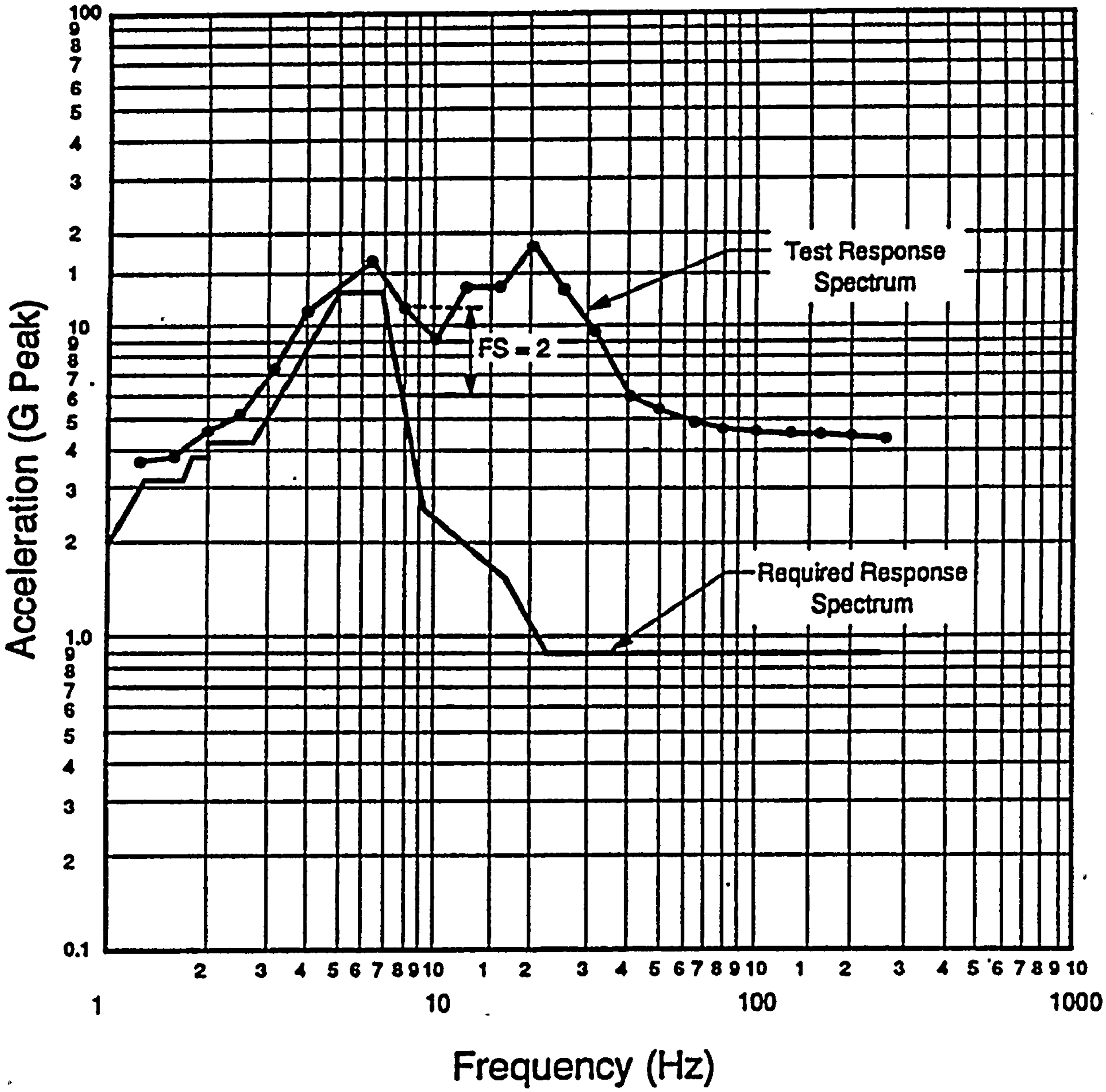

Figure 3.2-3
Typical Overtest at High Frequency

3.3 Analysis of Plant System and Structure Response

To complete the SPSA for WNP-2, evaluations of the response of Systems, Structures, and Components (SSCs) to postulated seismic events were completed. This section summarizes those evaluations.

3.3.1 Evaluation of Civil Structures

WNP-2 civil structures housing essential equipment include the containment vessel, containment internal structure (principally the reactor pedestal and sacrificial shield wall), reactor building, radwaste/control building, turbine building, diesel generator building, and standby service water pumphouse. These structures are expected to have substantial seismic capacity since (1) they are generally constructed of heavy reinforced concrete shear walls and floor slabs and (2) they were designed for seismic loads obtained by dynamic analysis for a Safe Shutdown Earthquake peak ground acceleration of 0.25g. Review and seismic evaluation of the civil structures was consequently performed only to verify that their HCLPF capacities are greater than 0.50g PGA. Seismic evaluation of the WNP-2 civil structures consisted of (1) initial review and (2) analysis of selected structural components. Initial review was performed to (1) obtain an understanding of the structure configuration, seismic load paths, and seismic responses and (2) select a subset of structural components for more detailed evaluation. Documents available included the structural drawings, Final Safety Analysis Report, other design basis information, and IPEEE seismic response analysis results. This information was reviewed to identify the primary seismic load paths within the structure, to qualitatively assess the structure seismic capacity, and identify any obvious seismic vulnerabilities. Based on this review, in conjunction with simplified, approximate calculations, individual structural components judged to be more heavily loaded in comparison to their capacities were selected for more detailed evaluation.

HCLPF capacities for the selected structural components were determined following the Conservative Deterministic Failure Margins (CDFM) method recommended in EPRI NP-6041-SL (Reference 3.6.2). Key features in the application of this method to the WNP-2 seismic IPEEE included the following:

- Conservative structure seismic demands were based on the 84% responses for a 0.50 PGA calculated in Reference 3.6.8. These overall structure seismic loads were distributed to the individual load-resisting members in proportion to their relative rigidities.

- Conservative structure seismic capacities were determined following acceptance criteria recommended by EPRI NP-6041-SL. These acceptance criteria are specified in appendices to EPRI NP-6041-SL as well as industry codes such as ACI 349-90, AISC Specifications, and Load and Resistance Factor Design (LRFD) specifications.. (Reference citations are included in Reference 3.6.5.)

- Following the recommendations of EPRI NP-6041-SL, an inelastic energy absorption factor of 1.25 was conservatively assigned to ductile failure modes in lieu of performing a more rigorous analysis.

Conservative approximations were frequently included to simplify the calculations. Higher seismic capacities could be obtained by additional analysis to reduce these conservatisms, if desired.

HCLPF capacities for the WNP-2 structures were calculated and are contained in EQE calculation notes. Complete details of these results are contained in Reference 3.6.5; summaries of the results are contained here:

Reactor Building

The reactor building houses the steel containment vessel and containment internal structures. At and below the refueling floor, the reactor building is constructed of reinforced concrete, with lateral seismic loads resisted by shear walls and floor slabs. The refueling floor is enclosed by a steel superstructure. The containment vessel is of welded steel construction and is founded on the reactor building base mat. The containment internal structures include the reactor pedestal, sacrificial shield wall, drywell floor, and stabilizer truss.

The steel containment vessel, stabilizer truss, and steel superstructure were screened from detailed evaluation. IPEEE seismic stresses in the containment vessel were noted to be relatively low. Review of the stabilizer truss found it to be of relatively rugged construction. Essential equipment are not supported by the steel superstructure. Its collapse capacity was judged to be greater than 0.5g PGA.

The following HCLPF capacities were obtained:

Structural Component	HCLPF Capacity
Overturning moment on east exterior wall at EL 422′-3″	1.1g
Overturning moment on biological shield wall at EL 422′-3″	0.51g
Shear on biological shield wall at EL 471′-0″	0.68g
Overturning moment on containment vessel outer skirt	0.89g
Overturning moment on containment vessel inner skirt	0.51g
Shear on sacrificial shield wall-reactor pedestal interface	1.1g
Overturning moment on sacrificial shield wall-reactor pedestal interface	0.93g
Shear transfer from drywell floor to containment vessel	0.50g

Simplifying conservatisms were included in each component evaluation. It was concluded that the HCLPF capacities of the reactor building, containment vessel, and containment internal structure were at least 0.50g PGA. Comments on structural components having lower seismic capacities follow.

The overturning moment capacity of the biological shield wall was estimated by applying provisions for evaluation of reinforced concrete shear walls to the cylindrical configuration. Shear capacity of the biological shield wall was based on the recommendations of Appendix N of EPRI NP-6041-SL.

The inner and outer containment vessel support skirts are anchored to the reactor building foundation by cast-in-place steel bolts. Following the provisions of ACI 349-90 Appendix B, anchorage capacities were found to be controlled by concrete pullout rather than anchor tension. Skirt overturning moment capacity was limited to the value at which concrete pullout initiates, and potential stress redistribution associated with ductile yielding was not permitted. An inelastic energy absorption factor of 1.0, applicable to brittle failure modes, was used. Overturning moment resistance provided by the weight of concrete foundation bearing on the skirt was conservatively neglected.

Tangential shears are transmitted at the interface between the drywell floor and the containment vessel. Capacity of the embedment detail in the drywell floor was based exclusively on the 7/8″ diameter Nelson studs. Additional shear capacity provided by embedded steel plates, which act as shear lugs, was conservatively neglected.

Radwaste/Control Building

The radwaste/control building is primarily a reinforced concrete structure, with lateral loads resisted by shear walls and floor diaphragms.

The following results were obtained:

Structural Component	HCLPF Capacity
Overturning moment on Column Line 15.1 wall at EL 467′-0″	0.51g
Shear on Column Line 10 wall at EL 501′-0″	0.50g
Shear on EL 501′-0″ diaphragm north of Column Line L.9	0.56g
Shear on EL 525′-0″ diaphragm west of Column Line 10	0.70g

Seismic demands on these components included contributions from very large torsional moments calculated by the IPEEE response analysis. Phasing of the torsional moments relative to the overall structure shears is unknown. The maximum shears and torsional moments were conservatively assumed to act concurrently.

The HCLPF capacity for the radwaste/control building was found to be at least 0.50g PGA. Additional comments on the seismic evaluation follow.

The overturning moment capacity of the wall on Column Line 15.1 accounted for available sources of resistance below the Elevation 467′-0″ slab, including continuous portions of the wall, cross-walls, and Column L.9-13.9. The capacity was determined following the approach for shear walls presented by Cardenas.

The shear capacity of the wall on Column Line 10 was derived from the recommendations of Appendix L of EPRI NP-6041-SL. Additional conservatism was included by factoring the contribution to shear strength from the steel reinforcement by $e^{-0.2}$ (0.819), which is the strength reduction factor applied to the concrete shear strength.

Numerous openings interrupt the load path for diaphragm shears being transmitted to the shear walls. Diaphragm shear capacities at these sections were evaluated using ACI code provisions for shear-friction.

Turbine Building

The lateral load-resisting system for the turbine building consists of reinforced concrete shear walls and floor diaphragms. The operating floor is enclosed by a steel superstructure. Limited quantities of equipment included in the seismic IPEEE SSEL are located in the turbine building.

Evaluation of the steel superstructure was not performed. There are no essential components attached to this part of the turbine building. Even if the superstructure were to collapse, it would not impact the reactor and radwaste/control buildings.

The following results were obtained:

Structural Component	HCLPF Capacity
Shear on Column Line 13 wall at EL 441′-0″	0.51g
Overturning moment on Column Line 13 wall at EL 441′-0″	0.60g
Shear on Column Line A wall at EL 441′-0″	0.59g
Overturning moment on Column Line A wall at EL 441′-0″	0.63g
Shear on diaphragm between Column Lines A and B EL 441′-0″	0.57g
Moment on diaphragm between Column Lines A and B EL 441′-0″	0.55g

As shown, the HCLPF capacity for the turbine building was found to be at least 0.50g PGA. Shear wall capacities were calculated as previously described.

Diesel Generator Building And Service Water Pumphouse

The diesel generator building and service water pumphouse are both reinforced concrete structures. Their overall lateral load-resisting systems were judged to have HCLPF capacities greater than 0.50g PGA since IPEEE seismic forces on these buildings are enveloped by demands on the other buildings.

The only potential seismic concern identified was the 10 inch thick reinforced concrete wall enclosing the room housing the diesel day tank. This wall is doweled to the floor slab at Elevation 441′-0″, but not the slab at Elevation 455′-0″. A clip angle anchored to the overhead slab provides top support for out-of-plane response away from, but not towards, the day tank.

The 10 inch wall was conservatively modeled as a vertical cantilever assuming that there is no restraint against out-of-plane translations at the top. Its out-of-plane seismic capacity was evaluated following the approach recommended in Appendix R of EPRI NP-6041-SL for lightly reinforced masonry walls, with the following modifications:

- Material properties and strength capacities applicable to the concrete wall were used.
- The wall displacement at failure was derived from rotational capacity criteria for reinforced concrete members in flexure contained in Appendix C of ACI 349-90.
- The inelastic energy absorption factor was based on the recommendations for concrete shear walls in Appendix M of EPRI NP-6041-SL. Because pinching of the hysteresis loops of the concrete wall is not as severe as that for masonry walls, this approach for evaluating nonlinear response is more appropriate than that recommended in Appendix R of EPRI NP-6041-SL.

The HCLPF capacity of the 10 inch concrete wall was calculated to be 0.77g.

Seismic-Induced Building Impact

The reactor, radwaste/control, and turbine buildings are separated from each other by gaps of 3 inches. The diesel generator building is separated from the reactor and radwaste/control buildings by gaps of 2 inches.

Preliminary calculations were performed to identify the buildings having the greatest potential for impact due to seismic-induced relative displacements. Potential impact between the reactor and radwaste/control buildings at Elevation 542′-0″ due to E-W seismic response was found to be the controlling case.

Maximum building displacements are not reported by the IPEEE seismic response analyses. Building displacements, d, were consequently estimated using floor zero period accelerations (ZPA) and fundamental structure frequencies, f, in cycles per second as follows:

$$d = ZPA/(2\pi f)^2$$

This assumes that structure response is dominated by the fundamental mode, which is the case for these buildings.

The maximum relative displacement between the reactor and radwaste/control buildings was conservatively estimated as the absolute sum of the individual building displacements. This is an upper bound value.

A maximum relative building-to-building displacement of 3.01 inches was calculated. This equals the reactor building-radwaste/control building separation gap of 3 inches. The HCLPF capacity for building impact was thus found to be 0.50g.

3.3.2 Evaluation of Equipment and Distributive Systems

Most commercial equipment and distributive systems are inherently rugged as long as they are adequately anchored. When this same equipment is qualified for seismic loading in a nuclear power plant environment, the qualification requirements and conservatisms in the specification of seismic demand and calculation of equipment response compound to result in very large margins. In Reference 3.6.2 screening guidelines are provided that, subject to a walkdown verification of the equipment to search for vulnerabilities, would allow most equipment and distributive systems to be screened for earthquakes up to 1.2g peak spectral acceleration. This value is to be compared to the 84th percentile ground motion spectrum defined at 5% damping. In Reference 3.6.5 screening thresholds were developed for equipment in terms of median and HCLPF capacities (see also Section 3.4, which follows). Based on the WNP-2 median UHS of Figure 3.2-2, a peak acceleration of 1.2g corresponds to a peak ground acceleration of 0.45g. The magnitude of the 84th percentile UHS is larger than the median. Therefore, the implied HCLPF of the screening tables of Reference 3.6.11 would be less than 0.5g required for screening based on the risk contribution criterion described in Section 3.4. The screening criteria in Reference 3.6.2 are based upon a large collection and review of earthquake experience data. The criteria also take into account the experience of performing many seismic PRAs in which the excess design margins have been quantified. In general, meeting the 1.2g screening criteria in Reference 3.6.2 would provide a good degree of confidence that the component is inherently rugged and would not fail in any earthquake that could potentially occur at WNP-2.

For the WNP-2 SPSA, the approach was taken that quantification of the conservatism in the design process must be used in conjunction with the walkdown, which is guided by the Reference 3.6.2 screening caveats, in order to screen out generic classes of components and distributive systems or individual components. Comparisons of the 0.5g median probabilistic floor response spectra to the original design floor response spectra for the SSE reveals that they are comparable. This comparison demonstrates that there is about a factor of two conservatism in the original seismic demand specified for equipment. This factor is less for rigid equipment but, since the spectral acceleration in the high frequency (rigid) regime of the floor spectra is much lower than for the amplified region, the loads for rigid equipment are also low. Comparisons of the SSE and 0.5g probabilistic spectra are made in Appendix B of Reference 3.6.5 for referral in the following discussions on screening derivations. The spectra comparisons alone do not demonstrate sufficient margin to screen out the components but they do make a significant contribution to the process.

Screening of Flexible Equipment and Distributive Systems Designed by Analysis

Flexible equipment and distributive systems which are designed by analysis usually have ductile failure modes but for purposes of screening it is assumed that the failure mode is non-ductile such as for failure of anchor bolts or welds or buckling. Fillet welds are shown in Reference 3.6.2 to have a much larger margin than implied by the design codes so fillet welds should not be the basis for the screening computation. The following assumptions are made for the screening calculation:

- Probable frequency range is 5-10 Hz (flexible equipment)
- 2% damping was used for the design whereas 5% is considered median with 2% defined as a -2 β_u case; (where β_u is the standard deviation)
- The screening is only applicable to locations outside of the primary containment vessel where the effects of hydrodynamic loads are minimal;
- Code margins are those inherent in the ASME code where the allowable stress may be as high as 70% of the specified ultimate strength, resulting in a nominal safety factor of 1.43. The safety factors of expansion anchors and welds are greater so the ASME criteria governs for the screening calculation.

The methodology used follows that described in Appendix A of Reference 3.6.5. Screening calculations are contained in calculations prepared by EQE.

Section 3.4 contains a summary discussion of the fragilities developed for the WNP-2 SPSA.

Equipment Inside of Containment

The FSAR contains summary information on the stresses or loads for critical elements of the primary system and many of the ECCS components. For those components outside of containment, the screening criteria developed for flexible and rigid equipment is applicable. Further review of the FSAR results and comparison of the design demand versus the probabilistic demand further confirmed the screening of these components.

For components inside of containment, including the reactor core support structure, further investigation was required. The original seismic design demand was based upon a stick model of the reactor building and containment and the spectra developed were very conservative as was the case for the Radwaste Building and Diesel Generator Buildings. During the resolution of the hydrodynamic loads issue, the Reactor Building was reanalyzed using a finite element soil-structure interaction model and the computed responses were much lower. This later analysis became the official design basis for equipment inside of the containment. Comparisons of the 0.5g median probabilistic spectra with the later finite element model spectra show that the probabilistic spectra are not generically enveloped by the design basis spectra, thus, the generic screening calculation employed for flexible equipment and distributive systems designed by analysis cannot be used to screen components inside of containment. For other equipment mounted in areas where the hydrodynamic loads from SRV or LOCA produce floor response spectra, the screening or fragility derivation must consider these effects.

Standard practice of General Electric is to design and analyze their scope of supply for envelope loads and confirm that the plant specific design loads/stresses are enveloped by the design loads/stresses. These envelope design loads are very conservative, especially when improbable load combinations of SSE and loss of coolant accident (LOCA) are included. Also, since the hardware was already purchased by the time the finite element model loads were developed, the hardware met the stick model response SSE requirements without hydrodynamic loads superimposed.

For the reactor vessel supports and the shroud support in the reactor internals, the design loads were compared to the loads from the 0.5g probabilistic response. The analysis of probabilistic spectra included the reactor vessel and internals model but forces were not computed at the RPV supports or for the internals so a substructure reactor vessel model was reconstructed and subjected to the median response spectra at the reactor vessel supports to obtain loads for comparison to the design loads. The analysis is contained as Appendix B of Reference 3.6.12. It was found that the faulted condition design loads were at least four times the 0.5g median probabilistic loads so a large margin exists for the reactor vessel supports and its internals. This is primarily because these components must be designed to resist direct jet force loads or annulus pressurization loads. The inertia loads from the building response spectra resulting from safety relief valve (SRV) discharge or from pipe break (LOCA) loads are much less significant than the direct forces. From this comparison we can also conclude that the design loads are very conservative for the recirculating pump and its supports and for the piping and valves from the reactor vessel and the recirculating lines.

In conducting the PRA it is assumed that there is no large pipe break in conjunction with the earthquake unless a weak link is discovered in the walkdown or qualification review. SRV loading is however a highly probable simultaneous occurrence. Typical spectra for SRVs and for the various events arising from pipe breaks were reviewed. The SRV spectra are significant in relation to the design earthquake loads and often as large as for LOCA loading (chugging). In design, the SRV loads are combined with the OBE and the allowable loads/stresses are held to much lower levels than for the SSE/SRV/LOCA load combinations so if the SRV loading is significant relative to the seismic loads, the OBE/SRV would govern the design and the SSE/SRV/LOCA would have a significant margin relative to allowable stresses for this load combination. If the design were governed by the SSE plus LOCA loads, the seismic plus SRV combination would also contain a significant margin.

The screening assumptions and criteria developed for the WNP-2 SPSA are summarized in Section 3.4.

Rigid Equipment

In general, rigid equipment has very high capacity due to its low seismic demand defined as the Zero Period Acceleration (ZPA) and its rugged features that result in it being rigid. Failures of rigid equipment is almost always due to failure of anchorage of very heavy massive equipment. For the most part, rigid equipment was screened out on the basis of the walkdown. Compact rigid equipment such as junction boxes, small pumps and tanks, chillers, small air handlers, etc. were in most cases screened. A generic fragility derivation for larger rigid equipment was developed based on the fragility methodology employed for flexible equipment incorporating the following conditions:

- Friction is included in the fragility calculation as a lateral load resisting mechanism duly accounting for the vertical acceleration.
- The location with the highest horizontal acceleration was used in the calculation. In this case, the contribution from friction is minimum.
- There is no conservatism from design damping since the spectral accelerations in the rigid range are equal for all damping levels.
- The uncertainty from modeling error is less since the component is not sensitive to frequency or mode shape error.
- Mode combination is not applicable for rigid mode response
- There is no conservatism or variability for response spectra peak broadening or smoothing in the rigid range

The result is a lower median with lower β as compared to flexible equipment.

Components Qualified by Test

Reference 3.6.4 provides a methodology for developing fragilities for components qualified by test. The median capacity is expressed as:

Am = (TRSC * FD * FRS * PGA)/RRSC (g)

Where:

TRSC = Clipped test response spectra

RRSC = Clipped required response spectra

FD = Device capacity factor

FRS is the structural response factor and

PGA is the peak ground acceleration for development of the RRS.

Continuing with the expansion:

TRSC = TRS (CT)(CI)

Where:

CT is set at unity for broad banded multiaxis testing

CI is a capacity increase factor and is recommended to be 1.1 with a β_U of 0.05 (Reference 3.6.4).

Also,

RRSC = RRS (CC)(DR)

Where:

RRS is the required response spectrum

CC is the clipping factor which is not applicable for the WNP-2 spectra since the spectral peaks are at very low frequency relative to the equipment fundamental frequencies.

DR is a demand reduction factor if the RRS is calculated by deterministic means. Probabilistic methods were used for the development of floor spectra so DR is unity.

FD is recommended in Reference 3.6.4 to be a value of 1.4 to demonstrate function during the earthquake with β_R of 0.09 and β_U of 0.22. Where β_R is the standard deviation due to randomness and β_U is the standard deviation due to uncertainty. FD for function after the earthquake is recommended to be 1.9 with β_R of 0.09 and β_U of 0.28.

FRS is the structural response factor which is unity for probabilistic spectra. The β_C is computed as the natural logarithm (ln) of the ratio of the 84th percentile response to the 50th percentile response at the frequency of the equipment, where β_C is the composite standard deviation from the following equation: $\beta_C = (\beta_C{}^2 + \beta_R^2)^{1/2}$. The βs vary with the location of the equipment. For purposes of the generic screening, an average ratio of 84th percentile to 50th percentile response of 1.25 was used and the β_C was computed to be 0.24 with β_R and β_U equal to 0.16 each.

PGA is the peak ground acceleration of the earthquake record used to develop the RRS. In this case the RRS was defined as the floor spectra for 0.5g median ground motion response.

Using the above methodology, the fragility may be expressed as:

Am	=	0.77 *(TRS/RRS)
β_R	=	0.18
β_U	=	0.28
HCLPF	=	0.36* (TRS/RRS)

If the TRS and RRS are equal at the fundamental frequency of the equipment the median capacity for function during the earthquake is 0.77g and the HCLPF is 0.36g. In order to screen out components with relays that must be functional during the earthquake, the TRS needs to be a factor of 1.6 greater than the RRS at the equipment fundamental frequency. This was easily demonstrated in most cases since the tendency is to overshoot the RRS for frequencies beyond that of the peak of the RRS (Figure 3.2-3).

Using the larger FD and larger uncertainty for function after the earthquake, the ratio of the TRS to RRS was computed to be about 1.3 for screening.

<u>Motor Control Centers</u>

Evaluation of MCCs was accomplished by combining field inspection and analysis results. Specifically, the analyses bridged the observer's variation between as-installed and as-tested MCC anchorage configuration. During the walkdown it was noted that the ITE MCCs were welded to channels which were in turn bolted to the concrete floor. The attachment of the channels to the floor appeared to be an order of magnitude stronger than the MCC/channel weld. Upon review of the qualification reports for the MCCs it was noted that the configuration tested was bolted instead of welded and it appeared that the tested condition was both stronger and stiffer than the welded configuration.

Yield line analyses and simplified non-linear analysis methods were performed using methodologies developed in References 3.6.4 and 3.6.16 to establish HCLPFs and fragilities for MCCs. The results of these analyses indicate that most MCC anchorages could be screened out. The exceptions were the MCCs at Elevation 525 in the Radwaste Building and at Elevation 411 in the Diesel Generator Building. The MCCs with the lowest fragility were determined to be E-MC-7F and 8F in the Radwaste Building at Elevation 525. The median capacity was compute to be 1.03g and a HCLPF of 0.44g for anchorage capacity and a median of 1.00g and a 0.43g for relay chatter.

3.4 Evaluation of Component Fragilities and Failure Modes

The methodology for evaluating seismic fragilities of structures and equipment is documented in Reference 3.6.4. Seismic fragility of a structure or equipment item is defined as the conditional probability of its failure at a given value of the seismic input or response parameter (e.g., ground acceleration, stress, moment, or spectral acceleration). Seismic fragilities are needed in a PRA to estimate the conditional probabilities of occurrence of initiating events (i.e., loss of emergency AC power, loss of forced circulation cooling systems) and the conditional failure probabilities of different mitigating systems (e.g., auxiliary cooling system).

The objective of a fragility evaluation is to estimate the ground acceleration capacity of a given component. This capacity is defined as the peak ground motion acceleration value at which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance capacity, resulting in its failure. The ground acceleration capacity of the component is estimated using information on plant design bases, responses calculated at the design analysis stage, as-built dimensions, and material properties. Because there are many variables in the estimation of this ground acceleration capacity, component fragility is described by a family of fragility curves; a probability value is assigned to each curve to reflect the uncertainty in the fragility estimation (Figure 3.4-1).

It is not practical to calculate fragilities for all components which are included in the risk modeling. Most components and distributive systems are inherently rugged and can be screened out on the basis that their seismic induced failure rate is low in comparison to the items which will ultimately dominate seismic risk. It is desirable to establish a fragility target above which components exceeding this target may be screened out.

In developing the target, three variables must be considered: (1) seismic hazard, (2) uncertainty in the median fragility and (3) frequency of failure (potential core damage) relative to that for other events. A fourth variable, consequence of failure is important, but for purposes of establishing a fragility cut off it is assumed that all failures have equal consequence. Parametric studies were conducted using the hazard and candidate fragility curves as input variables and examining the resulting failure frequency. The example fragility descriptions were convolved with the seismic hazard using the computer code EQE SRA to compute seismic failure rates. The example cases were then studied to determine an acceptable cutoff fragility level.

(1) Seismic Hazard: The seismic hazard is discussed in Section 3.1.1. Uniform hazard spectra are defined which describe the spectral accelerations at different frequencies for different return periods. The peak ground acceleration vs frequency of occurrence is provided up to 1.0g. NUREG-1407 states that the seismic hazard must be carried out to 1.5g unless sensitivity studies can show that a lower cutoff is justified. In the fragility cutoff study, the pga hazard was extrapolated to 1.5g and cases were run for 1.0 and 1.5g cutoff. For low capacity components, the extension of the hazard beyond 1.0g does not make a significant difference in calculated failure rate. However, at the fragility level that was ultimately determined to be an acceptable cutoff (about 1.5g median capacity), there was enough difference between the 1.0 and 1.5g cutoff results that the cutoff decision was based on a 1.5g pga cutoff for the hazard.

(2) Uncertainty in the Median Fragility: The uncertainty range for fragilities varies with the failure mode. For ductile modes of failure, such as for structures or piping, the margin to failure relative to code allowables is larger than for brittle or functional failure modes but the uncertainty is also larger so that a dual criteria must be implemented to establish a minimum value of the median capacity and of the HCLPF.

HCLPF is an acronym for high-confidence-of-low-probability-of-failure. It is defined mathematically as 95% confidence of less than 5% probability of failure. If the fragility curve is described by the median, Am, the randomness, β_R, and uncertainty, β_U, where the βs are logarithmic standard deviations, the HCLPF may be computed from:

$$HCLPF = A_m e^{(-1.65)(\beta_R + \beta_U)}$$

For ductile failure modes of flexible systems, such as for structures, the ratio of median to HCLPF is typically about three. For brittle failure modes of rigid equipment or functional failure modes such as relay chatter, the ratio of median to HCLPF tends to be closer to two. Thus, for the same seismic failure rate, the flexible, ductile items must have a higher median but may have a lower HCLPF than for a non-ductile failure mode.

The cases conducted to determine the fragility level for screening revealed that the seismic failure rate is more sensitive to HCLPF than median. Often it is more convenient to estimate or compute a deterministic HCLPF for making decisions on screening. The final cutoff value for fragility was targeted to a HCLPF value, wherein the median value is implied, depending upon the failure mode. Establishment of a HCLPF above the cutoff target was the approach used exclusively for screening of structures.

(3) Failure Rate Relative to Other Event Failure Rates: Core damage from internal events usually governs the plant risk. Internal event core damage frequencies (CDFs) typically are on the order of 1.0E-5/year. In order to prevent screening out seismic failures the could contribute more than 10% of the internal events CDF, this analysis set a target for screening of 1.0E-6 or less for the seismic failure rate for a 1.5g hazard cutoff.

Applying the logic described above and the results of the several analyses conducted, the following approximate fragilities were determined to be the threshold for screening out components:

Characteristics of Item	Am	HCLPF
Flexible, ductile structures or equipment	1.5	0.5
Brittle or functional modes of failure	1.22	0.57

These two cases result in seismic failure rates of approximately 1.0E-6 for a 1.5g cutoff. These values were used as surrogate fragilities for system capacity in seismic accident sequences where all components in a system are screened out. In this manner the failure rate of the screened out components is included in the seismic risk analysis.

Section 3.3 contains a summary of the methodologies employed for screening of SSCs, and for developing fragilities for those SSCs which could not be screened. References 3.6.5, 3.6.9, and 3.6.10 contain a detailed description, and the full results, of the fragility analyses employed for the WNP-2 SPSA. Reference 3.6.5 and 3.6.10 list the results of the screening and specific fragilities for the equipment on the SSEL. Reference 3.6.9 provides a refined analysis for the weaker components. Several notes are included that explain the basis for screening or for the tabulated fragility. Most components are screened out; the weakest links in the plant are due to anchorage failure and function of a small number of MCCs.

Flexible Equipment and Distributive Systems Designed by Analysis

For these types of equipment and systems, the fragilities were determined and screened as follows.

The Strength Factor, Fs, is:

$$Fs = (FU\text{-}N)/SSE$$

FU is the ultimate strength (stress/load), N is the normal load or stress and SSE is the seismic load or stress. Conservatively, consider that the only normal load effect is from weight which usually has an inconsequential effect on seismic capacity, thus the median normal load is assumed to be zero with a $(+1\beta)$ value equal to 15% of the ultimate capacity. The median strength is about 1.2 times the code value; the code value is set at the 95% confidence level, which is a $(-1.65\ \beta_U)$ value. It is assumed that the average demand is 70% of the code allowable with 100% assumed as a 95% probability value $(+1.65\ \beta_U)$. The code allowable is as high as 70% of the code ultimate strength, therefore the SSE load/stress is (.7)(.7) = 0.49 of the ultimate code capacity. The strength factor is then:

$$Fs = (1.2-0)/0.49 = 2.49$$

Using the approximate second moment method from Reference 3.6.4 for calculating β, the β_U is computed to be 0.29.

The equipment response factor consists of the product of the individual factors for the variables of Qualification Method, Damping, Modeling, Mode Combination and Earthquake Component Combination.

Typically dynamic response spectrum analyses were conducted as opposed to the simple, but conservative, static coefficient method. There is no particular bias in the response spectrum method but considerable conservatism can accumulate from several variables. The practice of peak broadening and smoothing introduces conservatism. Plant specific studies were not conducted for WNP-2 to quantify this conservatism. A prior study of single degree of freedom systems revealed that the degree of conservatism for spectra smoothing in the frequency range of interest is about a factor of 1.2 with a β_U of 0.09.

The factor of conservatism that results from using 2% damping in design vs 5% median is quantified by the ratio of the respective spectral accelerations:

$$FD = (Sa2\%)/(Sa5\%)$$

The reconstituted design spectra were used to find the average value of 1.17 for the spectral acceleration ratios between 2% and 5% damping in the 5-10 Hz frequency range. If 2% damping is a ($-2\ \beta_U$) value, the β_U for damping is 0.08.

Modelling error can arise from frequency error and mode shape difference between the model and the actual response. The model would normally be median centered so the modelling factor would be unity. The spectra peak at very low frequency are fairly flat between 5 and 10 Hz. Assuming a 1β frequency error of 15%, the β_U on response due to frequency error is conservatively estimated at 0.1. The response β_U due to mode shape error is estimated to be about 0.15. Combining βs by the square root of the sum of the squares (SRSS) method, the β_U for modelling is 0.18.

Mode combination was by SRSS which is median centered. The response variability due to mode combination is estimated by a β_R of 0.15.

For some commodities and civil structures, seismic components were combined by the absolute value of the worst horizontal plus the vertical component in design. If both horizontal components significantly contribute to component failure and the vertical earthquake contribution to component failure is small, this combination can be non-conservative relative to the SRSS or 100-40-40 rule.[3] Assume that the contribution to failure of one horizontal component is 2/3 of the other and the vertical contribution to failure is 10%. The vector of this combination is SRSS of $(1^2 + 0.67^2 + 0.1^2) = 1.21$ as opposed to the design value of $(1 + 0.1) = 1.1$. The safety factor for earthquake component combinations (F_{ECC}) is then $1.1/1.21 = 0.91$. Assuming all three components are in phase results in a $3\beta_R$ case, the β_R is 0.21.

[3] The 100-40-40 rule is suggested in NUREG/CR-0098, where the best (median) estimate of response to 3 directions of earthquake input is 100% of the dominant direction plus 40% of the other two orthoginal direction responses.

Combining the response factors as the product of the individual factors and the βs by the SRSS rule, the equipment response factor and its variability are represented by:

$$F_{RE} = 1.28$$
$$\beta_R = 0.26$$
$$\beta_U = 0.22.$$

Spectra were developed by probabilistic methods using a Latin Hypercube simulation process in which all important variables associated with structural response are included. The median results were used to derive the strength and response factors so the structural response factor is unity. The difference between the 50th and 84th percentile spectral accelerations in the 5-10 Hz frequency range defines the composite variability, β_C. This ratio averages about 1.25 for the reactor and radwaste buildings so the β_C is 0.22. There is approximately equal variability from random and uncertainty variables and the corresponding β_R and β_U are (0.24)1/2 = 0.16 each.

The median peak ground acceleration capacity is the product of the strength, equipment response and structural response factors times the reference 0.5g peak ground acceleration.

$$Am = 2.45\ (1.28)(1.0)(0.5) = 1.57g$$

The random and uncertainty variability are the SRSS of the β_Rs and β_Us for the three variables and are computed to be:

$$\beta_R = 0.31$$
$$\beta_U = 0.40$$

The HCLPF is computed as:

$$HCLPF = 1.57g\ e^{[(-1.65)(\beta_R + \beta_U)]} = 0.49g$$

This is equivalent to the 0.5g HCLPF and 1.50g median set as the screening threshold (described previously in this subsection) and, considering the conservative assumptions used in the derivation, this class of component and distribution system can be comfortably screened out subject to a walkdown verification that there are no apparent vulnerabilities. This calculation is used to screen out piping, cable trays and valves as well as anchorage of passive instrument racks or electrical distribution cabinets.

Valves are rigid but the piping systems in which they are mounted are flexible and the piping response dictates the demand for the valves. Valve qualification data were reviewed and the valves usually had a large design margin above the specified demand so the above derivation is also considered applicable to valves with the exception of those identified during the walkdowns that appeared to be outside of the seismic experience database. The MSIVs and certain self contained hydraulic operated valves were identified that warranted further evaluation. The qualification data for the MSIVs were reviewed and the margin was found to be sufficient to screen them out. The self contained hydraulic valves noted on chilled water systems were subsequently determined to be non-safety. The exception was for locations on the hydrogen recombiners. The valves in this case had supports on the operators so these were also screened out.

Equipment Inside of Containment

Section 3.3.2 discusses the approach used to address equipment inside of containment. Using generic fragility derivations such as shown in Section 6.1 of Reference 3.6.5, the following screening assumptions and criteria were formulated:

- A conservative estimate of the SRV loads and the SSE loads was obtained from the FSAR summaries by comparing the upset and faulted load combination resulting loads/stresses.
- The original design, before addressing new hydrodynamic loads, was based upon the stick model seismic results or the static seismic design load factors were generic and conservative relative to the SSE loads obtained from the stick model analysis.
- If the loads/stresses from the FSAR have a margin relative to the code allowable equal to or greater than the ratio of the 0.5g 50 percentile spectral acceleration to the stick model SSE spectral acceleration in the probable frequency range of interest, the component may be screened out.

This analogy, along with the stated margins relative to the design load combinations in the FSAR, was used to screen out many components. In many instances we also reviewed the specific qualification packages to verify the screening.

Appendix C of Reference 3.6.12 documents the design margins obtained from the FSAR and flags those components which are screened out on this basis.

Rigid Equipment

Section 3.3 discusses the conditions employed to derive the fragilities for rigid equipment. The generic value calculated for a bounding case is:

$$
\begin{aligned}
Am &= 1.05g \\
\beta_R &= 0.22 \\
\beta_U &= 0.30 \\
\text{HCLPF} &= 0.45g
\end{aligned}
$$

This generic value does not quite meet the screening criteria summarized at the beginning of this subsection, but it is close. Based upon the fact that this is a bounding case and in most locations the capacity would be higher, and based upon the walkdown observations, most rigid equipment were screened out. The exceptions were the diesel generator sets and the large air handlers. The anchorage of the air handlers looked more marginal than for other rigid equipment and the qualification documentation was reviewed to confirm that the margins relative to design code requirements were sufficient for screening. Because of the inability to screen all anchorage visually for the diesel generators, the qualification documents were reviewed to confirm that they could be screened out.

Summary of Fragility Results

Table 6-1 of Reference 3.6.5 lists the results of the screening and specific fragilities for the equipment on the SSEL. Table 3.5-1 presents the fragility curve information for all components which did not pass the screening criteria. For components where a dual fragility is provided, the first value is for function during the earthquake and the second value is for survival and function after the earthquake. Most components are screened out; the weakest links in the plant are due to anchorage failure and function of a small number of MCCs and for function of relays in control room cabinets.

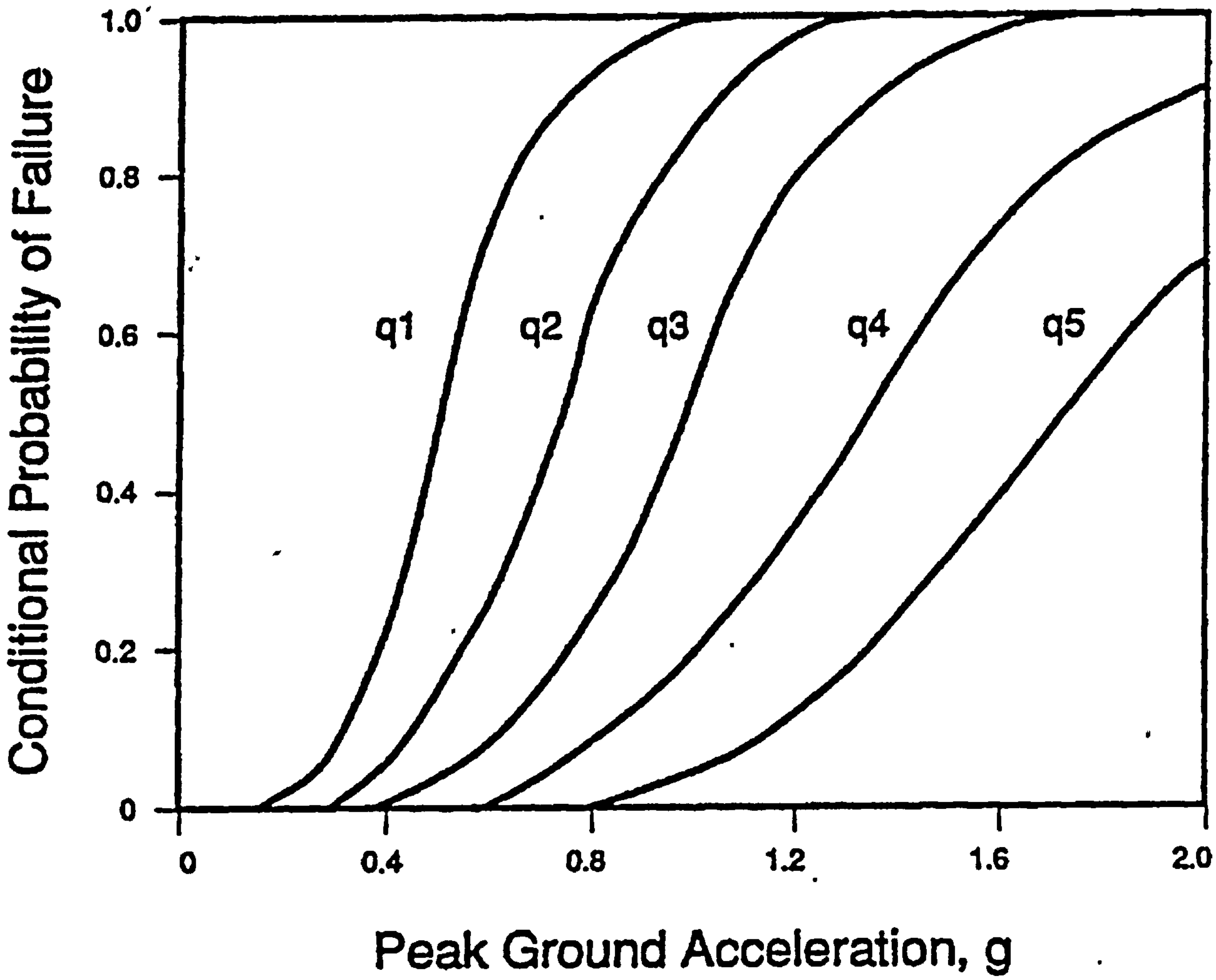

Figure 3.4-1
Typical Family of Fragility Curves for a Component

3.5 Analysis of Plant Systems and Sequences

The seismic probabilistic safety analysis (PSA) analysis was performed using the WNP-2 IPE model fault trees as the starting point. System logic and dependencies for normal plant operations were already built into this plant model. These fault trees were modified to include basic events representing seismic failure modes. Also the human error rates were modified to reflect a higher stress level which may be present in a seismic event. Seismic event trees were developed and the core damage equation was produced using the NUPRA computer code. The core damage equation was then quantified, convolving equipment fragilities and random failures with the hazard curve using the EQESRA computer code. In addition to quantifying the model as a whole, individual sequences were quantified to determine relative contributions to the core damage frequency and the NUPRA code was utilized to approximate the convolving process for the purpose of determining importance ranking of plant equipment in a seismic event. Each of these steps is explained in further detail in the sections which follow.

3.5.1 Fault Tree Development

3.5.1.1 Functional Effects of Unscreened Equipment and Coupling

Section 3.4 describes the process of developing screening levels and fragility curves for components at WNP-2. Those systems, structures, and components not screened by the seismic screening process at the review level earthquake (RLE) are presented in Table 3.5-1. In many instances, several component failures lead to the same system effect. For instance a major motor control center may feed several other smaller MCCs. If it is determined that the parent MCC is more likely to fail at a given acceleration level than the MCCs downstream (i.e., the parent MCC is the weakest link), only the failure mode of the parent MCC would be considered. In general, if a group of components is identified to perform a similar function and failure of any one of them would have the same result, the weakest component in the group in terms of ruggedness will be used to represent failures of the system, train or subsystem. To follow is a discussion of the unscreened equipment and the fault tree modifications made to model the seismic failures.

The liquid nitrogen storage tank located outside at the southeast corner of the diesel generator building did not pass the 0.5g RLE screen. The HCLPF for this component was calculated to be 0.42g. This component will be modeled by a basic event in the CN2 fault tree. There is a possibility that failure of this tank could cause oxygen starvation of the diesel generators whose intake structures are in the same vicinity. The Supply System studied this issue in 1992 (Ref 3.6.15) and concluded that in most cases the diesel generators would function well enough to carry design loads. The maximum time the diesels would be unavailable in the worst possible combination of weather conditions and puncture size is 8 minutes. This improbable delay in diesel function was not modeled in the seismic PSA.

E-SM-7, E-MC-7AA, E-MC-7F, E-CP-DG/CP1, E-CP-DG/REP1, E-CP-DG/RP1, E-SM-DG1/7, and E-SM-7/75/2 are all part of the Division 1 AC power distribution system. Of these components, E-MC-7F is the weakest, with a HCLPF of 0.43g for relay chatter and 0.44g for cabinet anchorage failure. If E-MC-7F fails, cooling to the Division I critical switchgear room is lost. This results in the failure of E-SM-7, E-SM-7/75/2 and the subsequent loss of the remaining components.

The Division 1 electrical components fall into three groups from a seismic point of view: 1) E-SM-7F is in a category by itself. When E-MC-7F is lost, cooling to the Division 1 critical switchgear room is lost. This results in the loss of Division 1 power. 2) E-SM-7, E-SM-7/75/2, and E-SM-DG1/7 all reside in the radwaste building and have identical fragility curves. Failure of E-SM-7 results in the failure of the other two panels. These three panels have fragility curves which are right at the screening level (HPCLFs are at 0.51g). In addition, the failure of these panels is enveloped by failure of E-MC-7F which has a lower seismic capacity. So, failure of these cabinets is enveloped both in effect and capacity and does not need to be modeled by an additional failure mode in the power distribution fault tree. 3) E-MC-7AA, E-CP-DG/CP1, E-CP-DG/REP1, and E-CP-DG/RP1 are all installed at the 441 elevation of the diesel generator bldg and have similar fragility curves. Of these four cabinets, the control panels have a slightly lower capacity, approximately the same as that of E-MC-7F. Like the E-SM-7 group, the effects of the diesel control panel failures are enveloped by the failure effects of E-MC-7F. However, because the diesel control panels are in a different building (experience different seismic input motion) and have approximately the same fragility, an additional failure point is added for these cabinets. This is done to model the fact that there will be some percentage of seismic events wherein the diesel will be lost but E-MC-7F will remain intact.

The seismic induced loss of division 1 electrical panels is modeled by two basic events in the EAC fault tree. One basic event which fails E-MC-7F at the E-MC-7F fragility curve levels and one basic event which fails DG-1 at the Diesel control panel relay chatter fragility levels.

E-SM-8, E-MC-8AA, E-MC-8F, E-CP-DG/CP2, E-CP-DG/REP2, E-CP-DG/RP2, E-SM-DG2/8, and E-SM-8/85/2 are all part of the Division 2 AC power distribution system. Of these components, E-MC-8F is the weakest, with a HCLPF of 0.43g for relay chatter and 0.44g for cabinet anchorage failure. If E-MC-8F fails, cooling to the Division II critical switchgear room is lost. This results in the failure of E-SM-8, E-SM-8/85/2 and the subsequent loss of the remaining components.

The Division 2 electrical components fall into three groups from a seismic point of view: 1) E-SM-8F is in a category by itself. When E-MC-8F is lost, cooling to the Division 2 critical switchgear room is lost. This results in the loss of Division 2 power. 2) E-SM-8, E-SM-8/85/2, and E-SM-DG2/8 all reside in the radwaste building and have identical fragility curves. Failure of E-SM-8 results in the failure of the other two panels. These three panels have fragility curves which are right at the screening level (HPCLFs are at 0.51g). In addition, the failure of these panels is enveloped by failure of E-MC-8F which has a lower seismic capacity. So, failure of these cabinets is enveloped both in effect and capacity and does not need to be modeled by an additional failure mode in the power distribution fault tree. 3) E-MC-8AA, E-CP-DG/CP2, E-CP-DG/REP2, and E-CP-DG/RP2 are all installed at the 441 elevation of the diesel generator bldg and have similar fragility curves. Of these four cabinets, the control panels have a slightly lower capacity, approximately the same as that of E-MC-8F. Like the E-SM-8 group, the effects of the diesel control panel failures are enveloped by the failure effects of E-MC-8F. However, because the diesel control panels are in a different building (experience a different seismic input motion) and have approximately the same fragility, an additional failure point will be added for these cabinets. This is done to model the fact that there will be some percentage of seismic events wherein the diesel will be lost but E-MC-8F will remain intact.

MC-7F and MC-8F are identical cabinets in similar locations within the radwaste building. They will experience similar seismic excitation and will respond to that input in a similar fashion. It is difficult to model this situation realistically. The HCLPF for these components is 0.43g[4]. For the purpose of this paragraph of discussion, the HCLPF will be simplified to imply a 5% chance of failure (rather than a 95% confidence of a 5% chance of failure). So, given a seismic event with a PGA of 0.29g, it is assumed there is a 5% chance that the motion will cause cabinet MC-7F to fail, and a 5% chance that it will cause MC-8F to fail. If the two failures were totally independent, the probability of both failing would be the intersection of these two probabilities or 0.25%. However, the two cabinets are very similar. They are of the same design, are loaded approximately the same, installation details are similar for the two cabinets, and their locations within the radwaste building are in different rooms but on the same floor, making the input motion they see very similar. Even with all of these similarities, an earthquake that topples one cabinet will not necessarily topple the other. But, the task of estimating the amount of coupling between the two cabinets is problematic. The probability that both will fail given a seismic event with a PGA of 0.43g falls somewhere between 0.25% and 5%. For this analysis, the two cabinets will be conservatively assumed to be perfectly coupled; when one fails the other also fails.

These cabinets will be coupled by using a single basic event identifier to model their failure. This basic event will be entered in the fault tree called "SINGLES" and will be utilized as an initiating event (see event tree section to follow).

The Div 1 and Div 2 diesel control cabinets are also identical between divisions and are mounted in similar locations. Failure of these cabinets result in the failure of the associated diesel generator to provide power to the bus. Based on the arguments presented above for MC-7F and 8F, it is also appropriate to couple the diesel generator control cabinets using a single basic event for both divisions.

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[4] The HCLPF calculations for these cabinets were updated by the Supply System (Reference 3.6.9) to remove undue conservatism. The resulting HCLPFs were 0.44g for relay chatter and 0.50g for anchorage failure. The corresponding fragility curves are represented by: A_m=1.05g, β_R=0.17 ,β_U=0.36 for relay chatter and A_m=1.18g, β_R=0.15 ,β_U=0.37 for anchorage failure.

E-SM-4, E-MC-4A, E-MC-4A/1, and HPCS-TB-D1008 are all part of the electrical distribution system which supports HPCS. Of these components, E-MC-4A/1 is the weakest with a HCLPF of 0.44. All four components are very similar in ruggedness. If E-SM-4 fails, E-MC-4A, E-MC-4A/1, and the equipment in HPCS-TB-D1008 all cease to provide any useful function. Thus, seismic induced loss of these components are conservatively coupled and modeled by a single basic event in the EAC fault tree which results in the loss of SM-4.

The switchyard at WNP-2 was judged to have a low seismic ruggedness, typical of switchyards worldwide. the HCLPF for the switchyard was estimated to be 0.10g. The seismic induced loss of offsite power will be modeled as an initiating event. See Section 3.5.2, event tree development, for further discussion.

Because of the low seismic capacity of the switchyard, it was decided early in the study that no credit would be taken for balance of plant systems in the seismic PSA. A seismic house event was added to several model fault trees to account for the unavailability of balance of plant systems. Specifically, the CST, TSW, and CAS are assumed to be unavailable. COND and RFW fault trees were not modified, but the systems are not credited in any seismic event tree.

Six safety related relays did not pass the RLE screening criteria. These relays are: E-RLY-52X/DG1B7, E-RLY-52X/DG1S7, E-RLY-SCR/DG1/7, E-RLY-52X/DG2B8, E-RLY-52X/DG2S8, and E-RLY-SCR/DG2/8. E-RLY-52X/DG1B7, E-RLY-52X/DG1S7, E-RLY-52X/DG2B8, and E-RLY-52X/DG2S8 can trip diesel generator loads requiring them to be resequenced if chatter occurs while the diesels are loaded. If E-RLY-SCR/DG1/7 or E-RLY-SCR/DG2/8 chatter while the diesel generators are operating they could cause the diesel generators to trip. The effects of these relay chatter events are bounded by the control panel failure considered above for panels E-MC-7AA and E-MC-8AA above. The failure of these panels makes the diesel generators unavailable.

3.5.1.2 Human Actions and Error Rates

The development of human error probabilities is included in Section 3.3.3 of the WNP-2 IPE submittal, revision 1. For the seismic analysis all operator actions which are required during a scenario have been conservatively updated to the levels appropriate for extreme stress. Human errors involving test and maintenance were not changed from the values calculated for the IPE submittal, because the test and maintenance activities are independent of the seismic event. Table 3.5-6 shows the SPSA human error failure probabilities and the associated IPE submittal failure probabilities for those basic events which were changed.

3.5.2 Event Tree Development

Of the seismic fragilities for components not passing the RLE screen, the most fragile component by far is offsite power. The fragility of offsite power is low enough that systems which require offsite power to operate were not reviewed to determine their fragility. For example, allocating resources to calculate the fragility of the condensate system would be wasted effort since without offsite power, condensate would be unavailable no matter how rugged it is. Clearly, the loss of offsite power should be modeled for the Seismic PSA. Because of the relative weakness of the switchyard, many other initiators considered in the internal events IPE do not come into play. The turbine trip initiator is not considered because the probability of tripping the turbine without offsite power is very low. MSIV closure, loss of feedwater, loss of condensate, loss of TSW, loss of CAS, and manual scram are similarly enveloped. Each of these initiating system losses would occur at seismic levels higher than loss of offsite power, and loss of offsite power results in the loss of all of these systems. Therefore, the seismic induced loss of power encompasses the listed initiators in scope of effects as well as exceeding them in frequency of occurrence.

Recovery of offsite power is not considered credible within the 24 hour mission time used for sequence quantification because the most probable failure mode in the switchyard is the failure of the ceramic insulators which support the power line. It is reasonable to expect large sections of the power lines to be down following large magnitude earthquakes. Therefore, recovery of offsite power and the systems mentioned above are not credited in the SPSA model.

Large and Medium LOCAs induced by earthquake have been screened out based on a thorough walkdown of the primary and connecting systems inside containment, and on seismic qualification data concerning the structures and components.

Seismic induced failure of small bore piping was also originally screened out based on the piping system design methods and detailed walkdowns. The walkdown identified no findings in which small bore piping in the primary or connecting systems appeared to be in jeopardy due to seismic action or seismic induced failure. However, NUREG 1407 clearly states that small break LOCA should be considered. Thus, although conservative for WNP-2, the small LOCA initiator was utilized in the seismic PSA. Because no specific instances of problems concerning small bore piping were encountered, no plant specific fragility is available. The screening fragility curve for elastic failure modes is utilized to model the small break LOCA initiator.

No mechanism was revealed to produce a seismic induced ATWS. Analysis of the vessel internal structure during the seismic fragility analysis performed by EQE International revealed a very conservative design. The design loads used for vessel internals are at least four times that produced by the 0.5g median probabilistic loads. Thus, control rod channel deformation causing an ATWS condition is not considered to be credible. The piping and components outside the vessel were also found to be very rugged with strengths exceeding the RLE loads. Therefore this event is not considered further.

The loss of Service Water (SW) initiator was not investigated explicitly because no specific vulnerabilities to seismic events were identified in the SW system. The supporting power system may be affected at levels below the RLE. However, the power distribution system is considered separately as an initiator.

The loss of Containment Nitrogen (CN) initiator was also not utilized. Though the liquid nitrogen storage tank was one of the few WNP-2 components that did not screen out at the RLE, loss of the tank does not constitute loss of the CN system. All vital functions served by the CN system have high pressure bottle backups. Thus, failure of the liquid nitrogen tank alone does not initiate a transient. The nitrogen backup sources and piping were inspected and found to have no specific seismic vulnerabilities based on the 0.5g RLE. The loss of the liquid nitrogen tank will be modeled in other sequences but not as an initiator.

The flooding initiators were also considered for seismic induced vulnerabilities. The suppression pool, condensate storage tanks, the hotwell, the spray ponds, the circulating water basins, the spent fuel pool, and the fire protection bladder tank were all walked down and studied for seismic vulnerabilities. All tanks other than the condensate storage tanks screened at the RLE. The condensate storage tanks are assumed to fail in a seismic event and be unavailable. The tanks are located outdoors and the ground is maintained to slope away from the buildings. Loss of the condensate storage tanks has an effect on the availability of HPCS and RCIC, but does not constitute an initiator.

Flood initiator category FLD7 in the internal events IPE was comprised of TSW piping and valves located in the reactor building. This section of the TSW system was covered in the seismic walkdowns and evaluations. The piping was found to be ruggedly built and supported, and screened at the 0.5g RLE. Flood categories FLD6 and FLD14 are comprised of CW, COND, RFW, TSW, and FP piping in the turbine generator building. Rupture of this piping causes flooding in the turbine generator building and can affect the COND, RFW, TSW, and CAS systems. These systems are a subset of the systems lost in the loss of offsite power event tree. As discussed above for the loss of COND, loss of RFW, the loss of TSW, and the loss of CAS events, the seismically induced loss of offsite power initiating event envelopes the seismically induced FLD6 or FLD14 initiators both in breadth of effects and in frequency of occurrence.

Loss of division 2 DC is considered as an initiator in the level 1 IPE. During the seismic evaluation both divisions were shown to be approximately equal in their vulnerability to seismic motion. The weakest link in the power distribution system was shown to be E-MC-7F and 8F. These MCCs provide power to the critical cooling systems for the division 1 and 2 switchgear rooms. Equipment in these rooms becomes unavailable quickly after loss of cooling. Recovery may be possible in the form of operator action to open the switchgear room doors and set up portable fans. However, as a first estimation, the loss of these two motor control centers was modeled as resulting directly in core damage.

Thus, the seismic PSA was modeled using 3 initiators: seismic induced LOOP, seismic induced small break LOCA, and seismic induced loss of power distribution. The last initiator providing a very simple event tree with one basic event.

3.5.2.1 Seismic Induced Loss of Offsite Power

Earthquake experience in the United States has shown that switchyards in general and power line support insulators specifically have rather low seismic ruggedness. The HCLPF calculated for the WNP-2 switchyard is 0.10g. Using this seismic capacity and the WNP-2 hazard curve it can be shown that the probability of LOOP due to a seismic event is a small fraction of the total LOOP initiating frequency. However, because of the possibility of additional seismic induced failures, the plant response can be significantly affected.

In general, the following sequence of events occur for the loss of offsite power initiator:

1. Recirculation pumps and condenser circulating water pumps trip off at zero seconds.

2. Due to loss of power to the SCRAM and MSIV relay solenoids, reactor SCRAM and MSIV closure is initiated at 2 seconds.

3. Feedwater turbines trip off at 4 seconds due to MSIV closure at 2 seconds.

4. Safety/relief valves open in the pressure relief mode of operation as the pressure increases beyond their setpoints.

5. Sensed reactor water level drops to the HPCS and RCIC initiation setpoint at approximately 36 seconds.

The event tree for the seismic induced loss of offsite power initiator is provided in Figure 3.5-1. The tree models the significant sequences involved in providing onsite power, maintaining core cooling and containment heat removal. The event tree and its functional events are discussed below.

E_E - Seismic LOOP Initiator

The fragility curve for seismic induced loss of offsite power is that calculated for switchyard damage. Discussion of fragility determination is covered in Section 3.4 of this report. Seismic failure in the switchyard is assumed to produce total loss of offsite power and is assumed not to be recoverable within a 24 hour time frame.

C - Reactor Subcritical

Failure to bring the reactor subcritical was not included in the development of the seismic induced LOOP tree since the frequency of such a scenario is less than 1E-7/yr. The likelihood for failure to SCRAM due to mechanical reasons was found in the WNP-2 IPE study to be 4E-6. The IPEEE analysis showed no additional seismic failure modes of the SCRAM system were credible. Convolving the switchyard fragility with the WNP-2 hazard curve yields a mean initiating frequency of 2.3E-4/yr. Thus, the combination of seismic induced LOOP and failure to SCRAM has a mean occurrence frequency of much less than 1E-7/yr.

DG - Either Diesel Generator 1 or 2 Available

In the event that offsite power becomes unavailable, AC emergency power can be supplied to the Division 1 and Division 2 4160 VAC buses via two independent diesel generators. The progression of events following loss of offsite power with failure to recover can be significantly different depending upon the availability of the emergency diesels. Successful diesel generator operation would make available several options for core cooling and containment heat removal, including LPCI, LPCS, and RHR. Additionally, DC power would be continuously supplied via the battery chargers allowing the continuous operation of RCIC and automatic ADS operation. With both diesels unavailable, a station blackout occurs. In this situation, HPCS can provide primary makeup if its dedicated diesel operates successfully. RCIC can also provide reactor coolant makeup for up to four hours prior to the depletion of station batteries. However, without recovery of either offsite power or the diesel generators, long term heat removal is unavailable, and core damage will result.

U_1 - HPCS Available

Following a loss of offsite power, Division 3 4160V AC emergency power can be supplied to HPCS via diesel generator 3. HPCS can provide reactor coolant makeup which is required in a relatively short time following a SCRAM and throughout the reactor shutdown period. HPCS can supply adequate inventory makeup to the reactor following a loss of offsite power and successful SCRAM. The HPCS main pump seals and bearings are cooled by its own discharge. However, the pump room is cooled by the Reactor Building Emergency Cooling using the SW-C train. Therefore, the Reactor Building Emergency Cooling, the SW-C and the HPCS must all be available for successful inventory makeup.

Referring to the LOOP event tree diagram in Figure 3.5-1, two unique functional equations were used for the U_1 function - one representing HPCS operation in LOOP conditions (U1-LOOP), and a second representing HPCS operation in station blackout conditions (U1-SBO). Since HPCS has a dedicated diesel generator, it would be expected that the solutions for U1-LOOP and U1-SBO would be nearly identical, as in fact they are.

U_2 - RCIC Available

RCIC is designed to start and run initially without any AC dependence. RCIC, although steam-turbine-driven, must rely on DC power (station batteries) for its operation. Following RCIC initiation, Loop B of the Standby Service Water system starts and the emergency cooling fan RRA-FN-6 of the RCIC pump room also starts if AC power is available. However, if AC power is unavailable, keeping the doors to the RCIC pump room open provides natural circulation sufficient to cool the room for continuous pump operation. See Section 3.5.1.2 for a discussion of human action failure rates used in the seismic PSA.

RCIC availability during a station blackout is strongly time dependent. This time dependence is principally due to the time varying auxiliary system requirements. The following considerations impact the availability of auxiliary systems required for RCIC coolant injection:

- Battery availability which is calculated to be 4 hours
- Room cooling requirements (RCIC system isolation on high room temperature by break detection logic)
- High suppression pool temperatures and containment pressure due to a lack of containment heat removal may have an adverse effect on the RCIC pump performance.

As with HPCS, two unique functional equations were generated for U_2: one representing RCIC operation in LOOP conditions (U2-LOOP), and a second representing RCIC operation in station blackout conditions (U2-SBO). The RCIC fault tree was solved to generate U2-LOOP by assuming the complete unavailability of offsite power, and U2-SBO by assuming the complete unavailability of AC power.

X - Timely Depressurization with ADS

In the unlikely event that there is an insufficient supply of coolant from high pressure sources, it would then be necessary for automatic or manual initiation of ADS to reduce reactor pressure below 470 psig to allow low pressure injection systems to maintain reactor inventory. WNP-2 has 7 ADS valves. Success criteria assumes that operation of 3 out of 7 ADS valves is required to depressurize the vessel so that the low pressure systems can be used. They will automatically actuate when the following signals are all present:

- Level 1 and Level 3
- 105 second time delay
- At least one low pressure ECCS pump is running.

The ADS may be actuated manually provided one low pressure ECCS pump is running. Operator failure to depressurize is analyzed in the human reliability analysis.

Since the reliability of depressurization is strongly sequence dependent, the reliabilities associated with automatic and manual depressurization may vary substantially with the sequence. Several items are of particular importance in the evaluated conditional probabilities:

- Battery power must be available to operate ADS solenoids.
- Automatic ADS would be inhibited in the case of a station blackout because of the low pressure pump operating requirement.

- Automatic ADS would occur if low reactor water level signals (L3 and L1) exist, one of the low pressure pumps is operating, Division 1 or 2, and the timer times out at 105 seconds.

- The WNP-2 emergency procedures direct the operator to depressurize if:

  a) Level cannot be maintained above
     -161" (TAF)
     -192" (2/3 Core Height)

  b) Suppression Pool (SP) temperature cannot meet heat capacity temperature limit

  c) Drywell temperature cannot be maintained below 340°F

  d) Wetwell pressure and level cannot be controlled below pressure suppression pressure limit.

- There is sufficient N_2 to operate the ADS valves. The CIA system supplies nitrogen to the 18 SRVs from the cryogenic nitrogen storage tank. This tank is one of the pieces of equipment which did not pass the RLE screening criteria and the failure of the tank is modeled using a fragility curve as well as with random failure modes. In the event cryogenic nitrogen is unavailable, two independent nitrogen bottle bank subsystems can deliver pressurized nitrogen to the 7 ADS valves and accumulators. A remote nitrogen cylinder connection is provided to each subsystem to permit supplementing the cylinder banks through manual connection of additional portable nitrogen cylinders, and thus maintaining pressure for an indefinite time.

- Without suppression pool cooling, the containment pressure may rise sufficiently to eliminate the required 88 psid differential pressure in the ADS pilot valves.

If ADS is unsuccessful in this sequence, no sources of reactor makeup are available, and core damage occurs.

The ADS functional event is not used for station blackout sequences due to the unavailability of low pressure reactor makeup systems. Note that although the automatic ADS function is not available, the operating staff can manually depressurize the vessel. In the event that diesel generator 1 or 2 is available, the ADS functional event is placed in the event tree for sequences in which the high pressure makeup systems, HPCS and RCIC, are unavailable. The unavailability of ADS was prepared using the ADS fault tree and accounts for the complete unavailability of offsite power.

V - Low Pressure Coolant Systems Available

The available low pressure coolant systems during a loss of offsite power are LPCS, LPCI, and FP water. The cross-tie from service water train B to RHR train B is assumed to be unavailable for simplicity due to the seismic failure events modeled for the SPSA, the most likely failure modes for RHR B involve loss of motive power which also cause the unavailability of SW train B. With the exception of the FP water pumps which are either motor-driven or diesel-driven, all low pressure pumps require 4160V AC power supplied by the diesel generators. Based on the fault trees developed for the low pressure systems, the unavailability for the low pressure systems is derived. The solution takes into account the complete unavailability of offsite power.

W_1 - Suppression Pool Cooling Available

The RHR system must provide a complete flow path from and to the containment (or reactor) through at least one RHR heat exchanger. In addition, the SW system must provide cooling water to the corresponding RHR heat exchanger from the spray pond. The RHR system has 2 loops for shutdown cooling mode, 2 loops for drywell spray mode, 2 loops for suppression pool spray/cooling mode, and 1 loop for vessel head spray mode. These modes are under manual control and are mutually exclusive. For IPEEE purposes, only the suppression pool spray/cooling mode is modelled. That model includes the RHR pumps, heat exchangers, and all of the salient valves. Although there are other RHR modes for removing decay heat, those other modes do not significantly add to the RHR availability. The solution of functional event W_1 in the E_E event tree takes into account the complete unavailability of offsite power.

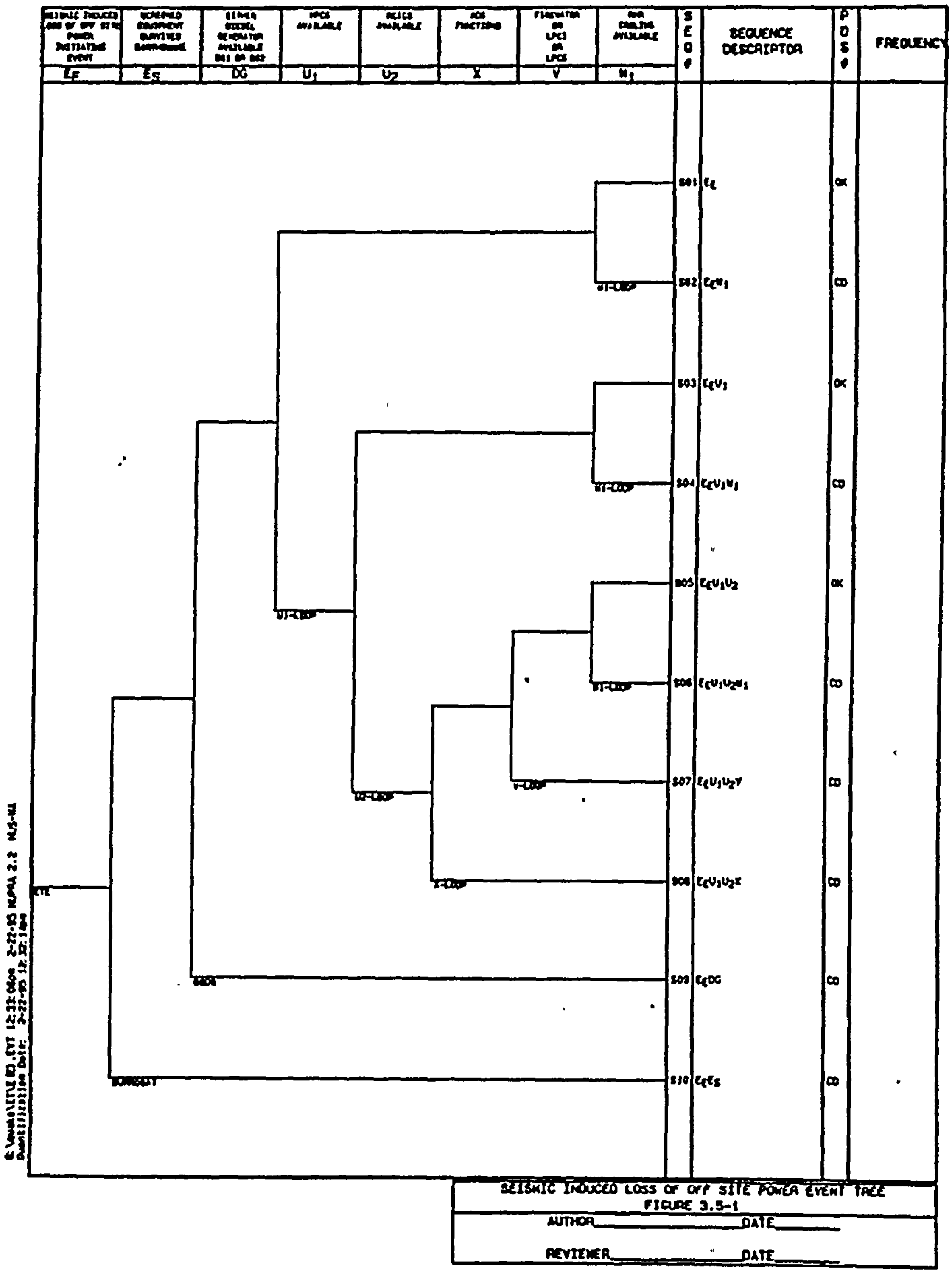


Figure 3.5-1
Seismic Induced Loss of Offsite Power Event Tree

3.5.2.2 Seismic Induced Small Break LOCA

Flow through a small break is a constant enthalpy process. If the primary system break is below the reactor water level, the blowdown will consist of reactor water. Blowdown from reactor pressure to drywell pressure will flash approximately one-third of this water to steam and two-thirds will remain as liquid. Both phases will be at saturation conditions corresponding to drywell pressure.

If the primary system rupture is located so that the blowdown flow consists of reactor steam only, saturated steam will result in superheated conditions. A small reactor steam leak will impose the most severe temperature conditions on the drywell structures and the safety equipment in the drywell.

After a small break in a pipe connected to the reactor vessel inside the primary containment, the vessel pressure and water level tend to slowly decrease, with a corresponding increase in drywell pressure and temperature. When the drywell pressure reaches 1.68 psig, a signal will be generated to SCRAM the reactor, start the diesel generators, and initiate the ECCS systems. For small breaks, the excess capacity of the feedwater system will compensate for the loss in vessel inventory due to break flow. Furthermore, the HPCS system will begin injecting water into the vessel. The operating staff will later take over manual control of the water makeup system to maintain level and proceed to a cold shutdown state.

RFW is assumed to be unavailable in the seismic IPEEE analysis. For the condition that HPCS is also unavailable, the reactor water level will continue to fall and finally reach the L2 trip setpoint. This trip will close the MSIVs, trip the recirculation pumps, and initiate RCIC. Once the MSIVs are closed, the reactor pressure soon rises to the SRV setpoint. The pressure then remains at basically the setpoint pressure as the SRV's cycle open and closed. The vessel pressure is maintained by steam generated by decay heat of the fuel.

The drywell pressure increase will lower the water level in the downcomer vents until the level reaches the bottom of the vents. At this time, noncondensibles and steam will start to enter the suppression pool. The steam will be condensed and the air will be carried over to the suppression chamber free space. The noncondensibles carryover will result in a gradual pressurization of the suppression chamber. Once all the drywell noncondensibles are carried over to the suppression chamber, pressurization of the suppression chamber will cease and the system will reach an equilibrium condition. The drywell will contain mostly superheated steam, and continued blowdown of reactor steam will condense in the suppression pool. The suppression pool temperature will continue to increase until the RHR heat exchanger heat removal rate is greater than the decay heat release rate.

The small LOCA event tree is shown in Figure 3.5-2. The event tree and its functional events are discussed below.

ES - Seismic Induced Small Break LOCA Initiator

As discussed in Section 3.5.2, the fragility curve used for seismic induced small break LOCA is the screening criteria fragility curve for ductile failures.

C - Scram

The combination of seismic induced LOCA and failure to SCRAM is not considered since the event frequency would be substantially less than 1E-7/yr. See Section 3.5.2.1 for further discussion. The seismic induced small break LOCA initiator frequency is lower than that for seismic induced LOOP. In fact, no credible seismically induced small LOCA mechanism was identified at WNP-2.

U - High Pressure Injection Availability

Either HPCS or RCIC can supply the high pressure injection requirements for a small break LOCA scenario. Further discussion of HPCS and RCIC availability can be found in Section 3.5.2.1 under the headings U_1 and U_2. The cutset equations from these two system fault trees are merged to provide the cutset equation under this heading.

For discussion of the remaining headings see Section 3.5.2.1 above.

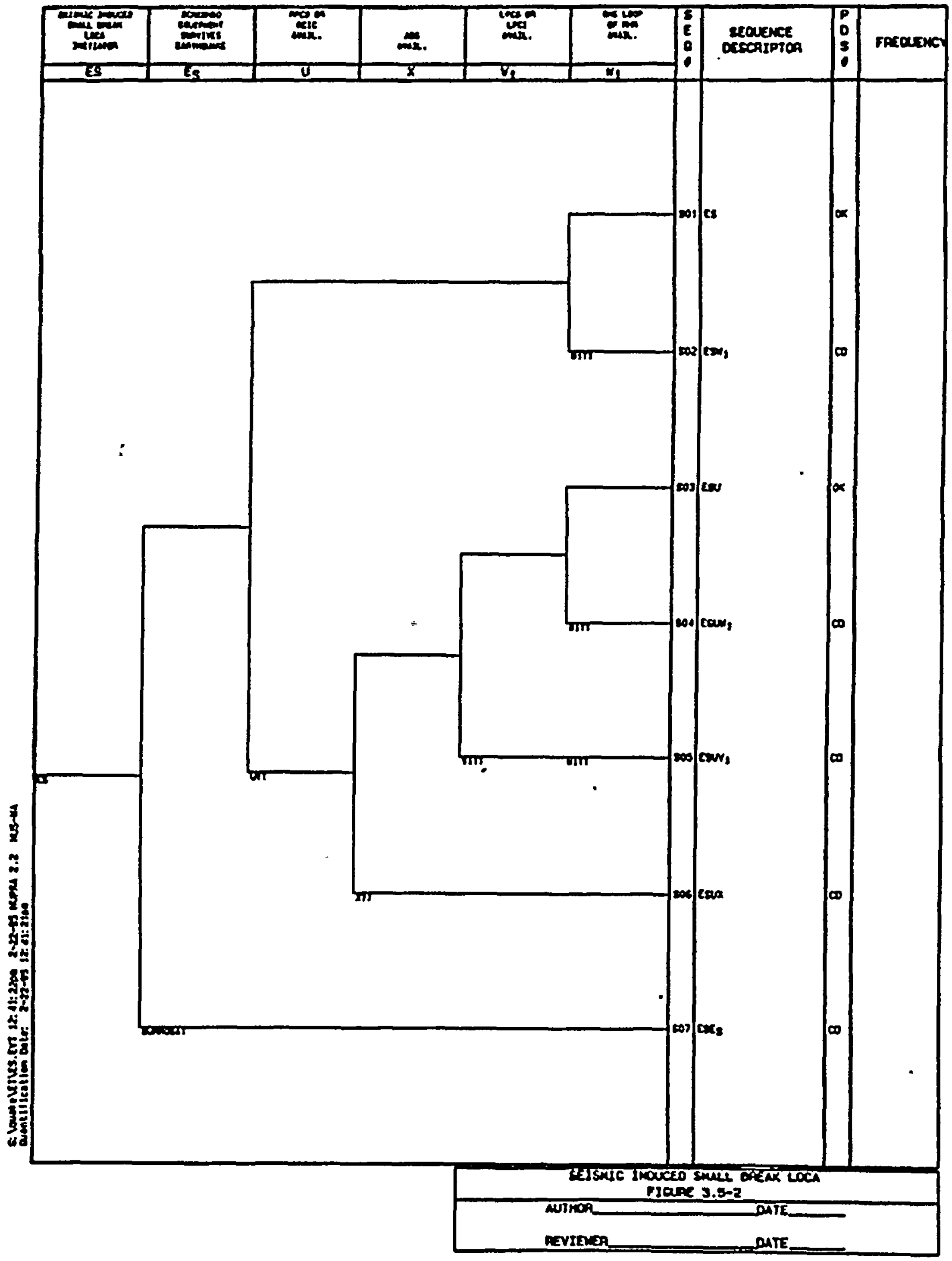


Figure 3.5-2
Seismic Induced Small Break LOCA

3.5.2.3 Loss of Critical Switchgear Room Cooling

Motor control centers E-MC-7F and E-MC-8F were found to have relatively low seismic capacity compared to other important equipment at WNP-2. These MCCs provide power for cooling the division 1 and division 2 critical switchgear rooms. The critical switchgear rooms have high thermal loads and heat up rapidly upon loss of cooling. The ability to SCRAM the reactor will be unaffected by this initiator. However, without restoration of the cooling to the division 1 and 2 switchgear rooms, the critical switchgear fail and power is lost to the low pressure injection and the containment heat removal systems. Without restoration of the power to these systems, core damage will ensue.

EM - Seismic Induced Loss of Critical Switchgear Cooling

The fragility curve for E-MC-7F and 8F is used for this initiator. The failure mode for these control centers is structural failure of the attachment to the floor.

REC - Recovery of Cooling

Recovery of the control centers is not considered credible in the available time frame. It may be possible for Operations to open the doors on each end of the switchgear rooms and set up portable cooling. However, no credit is taken for recovery of this scenario in this analysis.

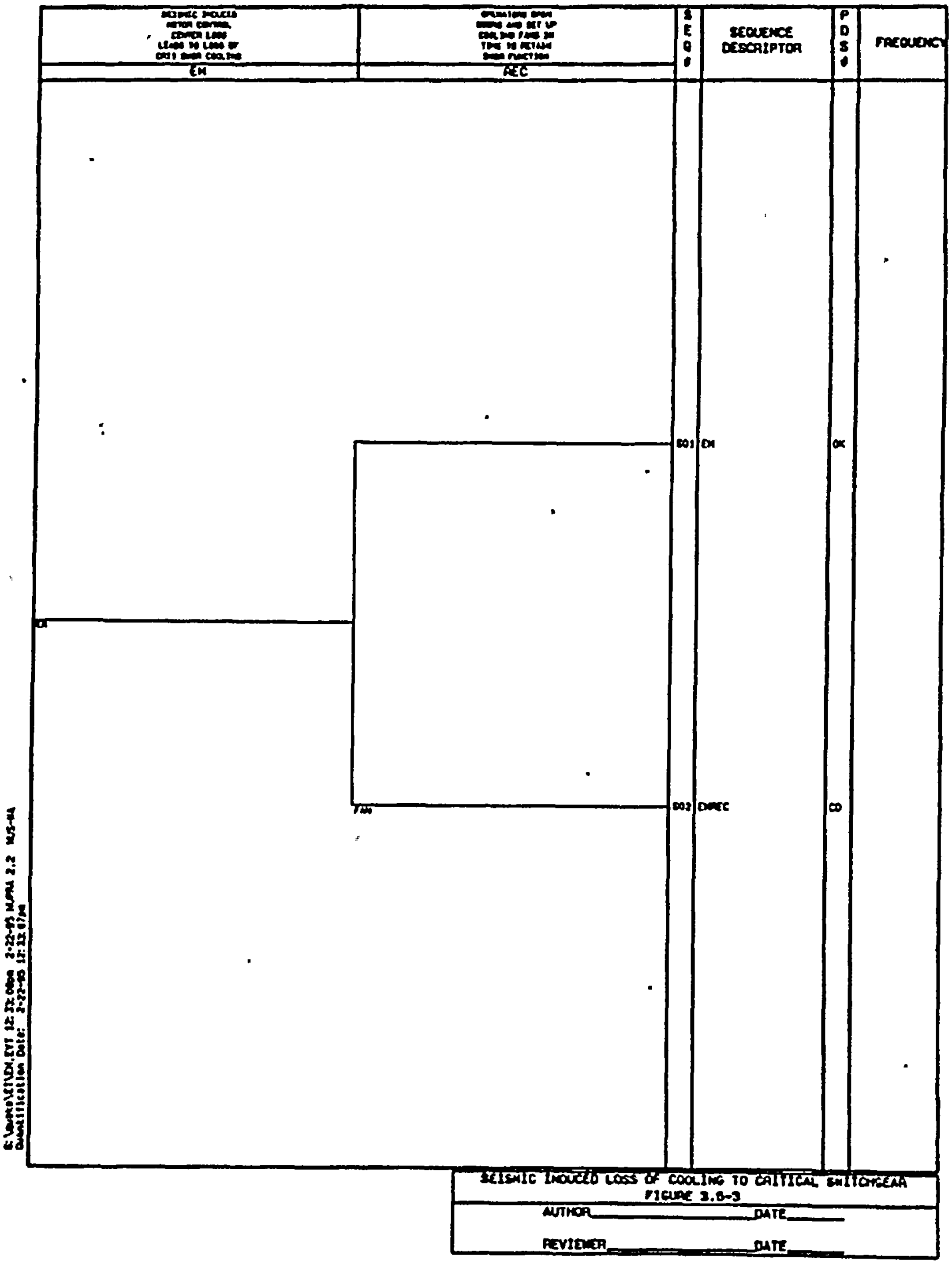


Figure 3.5-3
Seismic Induced Loss Division 1 and 2 Power

3.5.3 Seismic PSA Solution Process

Solving the seismic PSA model was a two step process. First the cutset equation was developed and minimalized using the NUPRA computer code developed by NUS. The cutset equation, including both random failures and seismic failure fragilities was then solved and convolved with the hazard curve using the EQESRA computer code developed by EQE International. In order to develop an equation that would include all of the important sequences for a seismic event, basic event failure rates, initiating event frequencies, and truncation cutoff values were carefully considered for the NUPRA runs.

For the cutset equation development, the seismic failure rate basic events were set to 1.0. Table 3.5-1 contains a list of the equipment for which a seismic failure mode was assigned. All of the human error rates which modeled operator actions during the event were raised to reflect a high stress environment. The initiation frequencies were originally set to 1.0. However, this resulted in a final minimalized cutset equation that was too large to quantify using EQESRA. Before the iterative process of raising the truncation limits was undertaken, careful consideration was given to the event tree initiating frequencies. Because loss of offsite power is substantially more probable in a seismic event than any of the other initiators, engineering judgement dictated that the initiating event frequencies be set to reflect this fact prior to truncation. The frequency for each initiator was set by taking the HCLPF acceleration and using the mean hazard curve to determine the corresponding frequency. This frequency was not meant to represent the actual initiating event frequency, but to build in a rational bias between event trees to allow the most important cutsets to overall core damage frequency to be retained in the final equation. Table 3.5-2 shows the initiating frequencies used in the event trees to develop the cutset equation.

At this point the NUPRA code was used iteratively to find the cutset truncation level that would result in a minimalized equation that was as large as possible yet would still run under EQESRA. The final truncation level was set at 1.2E-9.

The EQESRA code utilizes the uncertainty parameters, β_R and β_U, to expand the fragility curve information into a family of fragility curves for each component (see Section 3.4). For the SPSA a family of 5 fragility curves was developed for each seismic basic event and initiator. The fractional confidence levels, or probabilities associated with the 5 curves were 4%, 32%, 34%, 26%, and 4%. These levels were based on EQE International experience and were developed by subdividing a standard log normal probability density function into approximately equal intervals. The area under the log normal curve for each interval was determined and utilized as the confidence level. The hazard curve was also descritized into a family of 5 curves using the same fractional confidence levels. Figure 3.1-2 shows the hazard curve family.

The EQESRA computer code develops a plant fragility based on both seismic fragilities and random failure events. The code requires as input the uncertainty associated with the random failure events in the form of the ratio of the 95% upper confidence level to the mean failure rate. This information was not fully developed for the original IPE submittal. A literature search revealed that typical failure rate uncertainties run in the range of 2 to 20 for nuclear power plant components, with the majority falling in the range of 3 to 7. For the SPSA quantification a factor of 10 was used to estimate the uncertainty for all random failure events. The only exception to this setting was in the case of events that were set to a failure rate of 1.0. Basic events that were set to fail were modeled as having no uncertainty.

Table 3.5-3 shows the family of total plant fragility curves developed using the EQESRA computer code. Convolving the family of plant fragility curves with the family of hazard curves resulted in a mean core damage frequency of 2.1E-5/yr. The median, and the 5th, 10th, 90th, and 95th percentiles on the CDF results are as follows:

Mean Seismic CDF	2.1E-5/yr
5th percentile	3.9E-6/yr
10th percentile	4.1E-6/yr
Median	1.4E-5/yr
90th percentile	5.0E-5/yr
95th percentile	5.1E-5/yr

3.5.4 Sequence and Component Importance

The thrust of the IPEEE effort is to better understand the plant and the importance of plant components, systems and human actions. In order to understand the major contributions two methods of importance determination were developed: 1) A method to determine the relative contribution of individual event tree sequences, and 2) a method to determine the Fussel-Vesely importance of individual components, actions, and seismic failure modes.

To determine the relative contribution to CDF of individual cutset sequences, the cutset equations for each event tree sequence were quantified separately using EQESRA. The results of this exercise are shown in Table 3.5-4. About 95% of the total CDF is contributed by four sequences. The largest single contributing sequence is seismic induced SBO. Seismic induced failure of critical switchgear room cooling is the second largest contributor. Seismic induced loss of offsite power with subsequent failure to establish suppression pool cooling is third. And, seismic induced loss of offsite power with failure of all equipment that was screened out is the sequence ranked fourth in importance.

In order to determine the importance of individual components, an approximation of the process of convolving the fragilities and the hazard curve was performed using multiple NUPRA runs. The mean hazard curve was plotted and subdivided into ten intervals. For each interval the occurrence frequency was determined. Each seismic basic event and initiating event mean fragility curve was also plotted. The mean failure rate for the center point acceleration of each hazard curve interval was determined from the fragility plots. Ten NUPRA runs were made, one for each hazard curve interval. For instance, for the hazard curve interval of 0.05g PGA to 0.10 PGA the center point of 0.075g PGA was used to determine the mean failure rate of each seismic event using that event's mean fragility curve. For the seismic initiating events, the fragility value was multiplied by the interval frequency to yield an initiating frequency representative of the event tree initiation for the 0.05g to 0.1g interval. These initiating and failure rates were then used along with the model random failure rates to solve the overall cutset equation for the 0.05g to 0.10g interval. For each interval, the Fussel-Vesely importance was calculated for all basic events.

For each component a frequency weighted Fussel-Vesely importance average was calculated. That is, for each component the importance for each interval was multiplied by the interval occurrence frequency, summed, and the total divided by the total frequency of all intervals.

This method roughly approximates the solution process provided by the EQESRA computer code. It does not account for the uncertainty in the fragility or hazard curves or the uncertainty in the random failure events. This process also produces conservative results at high PGAs. The reason for this conservatism is that as the PGA increases, the fragility curve yields a failure probability close to 1.0. When NUPRA combines several terms with failure rates approaching 1.0 it yields event tree branch failure rates in excess of 1.0. EQESRA properly treats the boolean combination of the top end of the fragility curves, so this conservatism does not show up in the total CDF or sequence quantification results. The end effect of the conservatism in the approximation method is that the final Fussel-Vesely importance measures are weighted toward very large seismic events. The result of the importance measures exercise is shown for the top 50 events in Table 3.5-5.

The three most important events are seismic failures; Failure of the Division I and Division II motor control centers, E-MC-7F and 8F; Failure of the emergency diesel control panels; and the basic event used to model all plant equipment that was screened in the seismic quantification process.

The most important random failure events carry a common theme of power distribution. Diesel failure modes and DC power distribution failures top the list for random failures. This is not unexpected since the most important sequence involves loss of offsite power. High pressure injection and long term heat removal failure modes are also shown to be important in seismic initiated events.

The most important human actions determined by the SPSA are the actions to start suppression pool cooling when required, and the action to initiate LPCI injection if it fails to initiate automatically.

3.5.5 Recovery Actions

No credit was taken for recovery actions in the SPSA. By taking credit for recovery, the CDF could be lowered. Recovery of the critical switchgear rooms after loss of cooling, recovery of random failure events for the emergency diesels in the LOOP sequences, and recovery of long term heat removal in sequences where RHR is out for test/maintenance are possible. However, these options were not explored because the final CDF did not warrant further effort.

3.5.6 Sensitivity Studies

3.5.6.1 NUPRA Quantification Truncation Limits

As discussed earlier in the calculation, the truncation limit had to be adjusted in NUPRA to produce a cutset equation that would not exceed the limitations of EQESRA. The final cutset equation was produced using a truncation limit of 1.2E-9. Two additional runs were made using truncation limits of 1.0E-8 and 1.0E-7 to study the importance of the truncation limit.

The cutset equation using a truncation limit of 1.0E-8 was evaluated using EQESRA and produced a mean frequency of failure of 2.06E-5. This is less than 4% different than the result produced using the truncation limit of 1.2E-9.

The cutset equation produced using a truncation limit of 1.0E-7 was evaluated using EQESRA an produced a mean frequency of failure of 1.77E-5. Thus 82% of the CDF remains when a truncation limit almost two orders of magnitude higher is used to produce the cutset equation. Thus, it is determined that the truncation limits imposed by the interface limitations between NUPRA and EQESRA do not materially affect the results of the calculations.

3.5.6.2 Random Failure Uncertainty

In the process of calculating the plant fragility, the computer code EQESRA folds the uncertainty in the random failure basic events into the result. As discussed earlier, random failure rate uncertainties were not established for each basic event in the WNP-2 safety assessment model. This uncertainty was conservatively assumed to be a factor of 10. To determine the effect of this assumption, the uncertainty of all random failure basic events was set to 3.0 and the cutset equation was evaluated again using EQESRA. This assumption turns out to be fairly significant. The results using an assumed factor of 3.0 uncertainty for all random failure events were:

Mean Frequency of Failure = 1.27E-5
5% Confidence Bound = 1.54E-6
95% Confidence Bound = 3.22E-5

This is about a 40% reduction in CDF. Thus, the uncertainty of the random failure basic events make a significant impact on the final number. However, the relative importance of the various failure modes remain the same.

Table 3.5-1

FINALIZED FRAGILITY CURVES FOR UNSCREENED EQUIPMENT							
EPN	Description	Bldg.	Elev.	A_m	β_R	β_U	HCLPF
CN-TK-1	Liquid Nitrogen Storage Tank	Outside	441	0.86	0.18	0.25	0.42
E-CP-DG/CP1	Diesel Gen 1 Control	Diesel	441	0.95	0.18	0.28	0.44
E-CP-DG/CP2	Diesel Gen 2 Control	Diesel	441	0.95	0.18	0.28	0.44
E-CP-DG/REP1	Exciter Control Panel DG-1	Diesel	441	0.95	0.18	0.28	0.44
E-CP-DG/REP2	Exciter Control Panel DG-2	Diesel	441	0.95	0.18	0.28	0.44
E-CP-DG/RP1	Diesel Gen Reg Control Panel DG-1	Diesel	441	0.95	0.18	0.28	0.44
E-CP-DG/RP2	Diesel Gen Reg Control Panel DG-2	Diesel	441	0.95	0.18	0.28	0.44
E-MC-4A	HPCS Motor Control Center 4A	Diesel	441	1.1	0.12	0.36	0.5
E-MC-4A/1	HPCS Motor Control Center 4A/1	Diesel	441	.99/1.17	.15/.12	.35/.36	.44/.53
E-MC-7AA	Div 1 Diesel MCC 7A-A	Diesel	441	1.19/1.56	.15/.12	.35/.36	.52/.71
E-MC-7F	Motor Control Center 7F	Radwaste	525	1.00/1.03	.17/.15	.36/.37	.43/.44
E-MC-8AA	Motor Control Center 8A-A	Diesel	441	1.09/1.19	.15/.12	.35/.36	.48/.54
E-MC-8F	Motor Control Center 8F	Radwaste	525	1.00/1.03	.17/.15	.36/.37	.43/.44
E-SM-4	HPCS 4160V Switchgear	Diesel	441	1.03	0.18	0.28	0.48
E-SM-7	Div 1 4160V Switchgear	Radwaste	467	1.1	0.18	0.28	0.51
E-SM-7/75/2	4.16KV SWGR for Breaker 7/75/2	Radwaste	467	1.1	0.18	0.28	0.51
E-SM-DG1/7	DG-1 4160V Switchgear	Radwaste	441	1.1	0.18	0.28	0.51
E-SM-8	Div 2 4160V Switchgear	Radwaste	467	1.1	0.18	0.28	0.51
E-SM-8/85/2	4.16KV SWGR for Breaker 8/85/2	Radwaste	467	1.1	0.18	0.28	0.51
E-SM-DG2/8	DG-2 4160V Switchgear	Radwaste	441	1.1	0.18	0.28	0.51
HPCS-TB-D1008	HPCS Relay Terminal Box	Diesel	441	1.05	0.18	0.29	0.48
Offsite Power	Switchyard	Outside	441	0.31	0.25	0.43	0.10

Table 3.5-2

Initiating Frequencies Utilized for Cutset Equation Development

Initiator	Description	A_m	β_R	β_U	HCLPF	Initiating Frequency
E_E	Seismic Induced Loss of Offsite Power	0.31	0.25	0.43	0.10	1.3E-3
ES	Seismic Induced Small Break LOCA	1.50	0.30	0.36	0.50	2.2E-5
EM	Seismic Induced Loss of Division 1 and 2 Distribution Panels	0.70	0.17	0.36	0.29	1.2E-4

Table 3.5-3

WNP-2 PLANT FRAGILITY CURVE FAMILY					
PGA	Curve 1	Curve 2	Curve 3	Curve 4	Curve 5
.050	1.051E-07	1.466E-07	2.709E-07	4.909E-07	1.141E-06
.100	1.715E-04	2.336E-04	4.108E-04	7.208E-04	1.625E-03
.150	1.524E-03	1.981E-03	3.102E-03	4.956E-03	1.025E-02
.200	5.040E-03	6.372E-03	9.305E-03	1.392E-02	2.687E-02
.250	9.649E-03	1.205E-02	1.707E-02	2.472E-02	4.591E-02
.300	1.437E-02	1.777E-02	2.464E-02	3.496E-02	6.325E-02
.350	1.921E-02	2.386E-02	3.289E-02	4.697E-02	8.119E-02
.400	2.358E-02	2.947E-02	4.082E-02	5.841E-02	2.256E-01
.450	2.827E-02	3.594E-02	5.130E-02	8.088E-02	4.598E-01
.500	3.362E-02	4.398E-02	6.652E-02	1.308E-01	6.942E-01
.550	3.934E-02	5.395E-02	9.708E-02	2.212E-01	8.574E-01
.600	4.534E-02	6.551E-02	1.599E-01	3.507E-01	9.439E-01
.650	5.242E-02	8.368E-02	2.563E-01	5.056E-01	9.808E-01
.700	6.253E-02	1.153E-01	3.760E-01	6.594E-01	9.942E-01
.750	7.643E-02	1.646E-01	5.036E-01	7.907E-01	9.984E-01
.800	9.853E-02	2.357E-01	6.273E-01	8.837E-01	9.996E-01
.850	1.321E-01	3.252E-01	7.373E-01	9.410E-01	9.999E-01
.900	1.754E-01	4.268E-01	8.269E-01	9.724E-01	1.000E+00
.950	2.264E-01	5.318E-01	8.940E-01	9.880E-01	1.000E+00
1.000	2.905E-01	6.325E-01	9.393E-01	9.951E-01	1.000E+00
1.050	3.658E-01	7.220E-01	9.673E-01	9.981E-01	1.000E+00
1.100	4.494E-01	7.971E-01	9.832E-01	9.993E-01	1.000E+00
1.150	5.369E-01	8.569E-01	9.918E-01	9.998E-01	1.000E+00
1.200	6.229E-01	9.023E-01	9.962E-01	9.999E-01	1.000E+00
1.250	7.026E-01	9.356E-01	9.983E-01	1.000E+00	1.000E+00
1.300	7.718E-01	9.591E-01	9.993E-01	1.000E+00	1.000E+00
1.350	8.295E-01	9.750E-01	9.997E-01	1.000E+00	1.000E+00
1.400	8.759E-01	9.853E-01	9.999E-01	1.000E+00	1.000E+00
1.450	9.117E-01	9.917E-01	9.999E-01	1.000E+00	1.000E+00
1.500	9.385E-01	9.955E-01	1.000E+00	1.000E+00	1.000E+00

Table 3.5-4

SEQUENCE CONTRIBUTION TO CORE DAMAGE FREQUENCY		
Event Tree Sequence	Brief Event Description	Percent Contribution to CDF
E(E)S09	LOOP with DG-1 & DG-2 failure	51.9%
EMS02	Failure of Critical Switchgear room cooling	21.8%
E(E)S02	LOOP with failure to establish suppression pool cooling	14.8%
E(E)S10	Loop with failure of all screened equipment	6.6%
E(E)S04	Loop with failure of HPCS and suppression pool cooling	2.3%
ESS07	Small break LOCA with failure of all screened equipment	1.1%
	All other sequences	1.5%

Table 3.5-5

FUSSELL-VESELY IMPORTANCE MEASURE		
Basic Event	Brief Description	Overall F-V Importance
EACMC--7F8F--3EQ	Seismic Induced Loss of Crit Switchgear Room Cooling	4.00e-01
EACCP-DIESEL-3EQ	Seismic Induced Loss of Diesel Generator Controls	3.19e-01
SURROGATE----3EQ	Seismic Induced Loss of all Screened Equipment	1.21e-01
EACENG-EDG-2S4D2	DIVISION 2 EMERGENCY DIESEL GENERATOR EDG-2 FAILS TO RUN FOR 10 HOURS	4.51e-02
EACENG-EDG-1S4D1	DIVISION 1 DIESEL GENERATOR EDG-1 FAILS TO RUN FOR 10 HOURS	4.31e-02
EACENG-EDG-2W2D2	DIVISION 2 EMERGENCY DIESEL GENERATOR EDG-2 FAILS TO START ON DEMAND	3.25e-02
EACENG-EDG-1W2D1	DIVISION 1 DIESEL GENERATOR EDG-1 FAILS TO START ON DEMAND	3.10e-02
RHRHUMNSP-COOLLL	OPERATOR FAILS TO INITIATE SUPPRESSION POOL COOLING OR SPRAY	1.73e-02
EACCB--CBB-8B2L2	4160 VAC CIRCUIT BREAKER E-CB-B/8 FROM E-TR-B TO E-SM-8 FAILS TO OPEN ON DEMAND	1.68e-02
RHRHUMNLPCISTART	OPERATOR FAILS TO INITIATE LPCI MANUALLY	1.63e-02
EACCB--CBB-7B2L1	CIRCUIT BREAKER B-7 FAILS TO OPEN PPM 7.4.8.1.1.1.2	1.60e-02
EACEDG123---C3LL	COMMON CAUSE FAILURE OF DIESEL GENERATORS EDG-1, EDG-2, AND EDG-3	1.41e-02
RHRBTM	TEST/SCHEDULED MAINTENANCE UNAVAILABILITY OF RHR TRAIN B	9.40e-03
RRAFC03	RRA-FC-03 RELATED COMPONENTS FAIL [MODULE]	1.84e-03

FUSSELL-VESELY IMPORTANCE MEASURE		
RRAFC10	RRA-FC-10 RELATED COMPONENTS FAIL [MODULE]	1.83e-03
RRAFC02	RRA-FC-02 RELATED COMPONENTS FAIL [MODULE]	1.81e-03
RRAFC11	RRA-FC-11 RELATED COMPONENTS FAIL [MODULE]	1.80e-03
EACCB-EDG1-2C2D1	COMMON CAUSE FAILURE OF EDG-1 AND EDG-2 OUTPUT CIRCUIT BREAKERS	1.59e-03
EACOP--SWYRD-3EQ	Seismic Induced Switchyard Failure	1.46e-03
SWBTM	TEST/SCHEDULED MAINTENANCE UNAVAILABILITY OF SSW TRAIN B	1.40e-03
SW-P-MDSWP1BR2LB	SSW PUMP SW-P-1B FAILS TO START ON DEMAND	1.31e-03
EAC-LOGIC4--W3D2	FAILURE OF FAST BUS TRANSFER LOGIC TO ENERGIZE E-SM-8 FROM EDG-2	1.28e-03
RHRP-MD---2BR2LL	RHR PUMP RHR-P-2B FAILS TO START ON DEMAND	1.27e-03
RHRP-MD---2AR2LL	RHR PUMP RHR-P-2A FAILS TO START ON DEMAND	1.24e-03
SW-P-MDSWP1AR2LA	SSW PUMP SW-P-1A FAILS TO START ON DEMAND	1.24e-03
EAC-LOGIC3--W3D1	FAILURE OF THE FAST BUS TRANSFER LOGIC TO ENERGIZE E-SM-7 FROM EDG-1	1.22e-03
EACCB--DG2-8B2D2	4160 VAC CIRCUIT BREAKER E-CB-DG2/8 FAILS TO CLOSE ON DEMAND	1.19e-03
EACCB--DG1-7B2D1	4160 VAC CIRCUIT BREAKER E-CB-DG1/7 FAILS TO CLOSE ON DEMAND	1.13e-03
RHRATM	SCHEDULED TEST/ MAINTENANCE UNAVAILABILITY OF RHR TRAIN A	1.03e-03

FUSSELL-VESELY IMPORTANCE MEASURE		
RHRRLY--K70BT3LL	TESTING OF RHR-RLY-K70A & B PPM 7.4.3.3.1.12	9.63e-04
SW-CHR-SCC1BW2LB	FAILURE OF SSW PUMP ROOM COOLING COIL PRA-CC-1B	9.18e-04
WMACHR---53BW2LL	FAILURE OF COOLING COIL WMA-CC-53B IN CRITICAL SWITCHGEAR ROOM	9.18e-04
DMACHR----21W2LL	FAILURE OF COOLING COIL DMA-CC-21 IN EDG-2 ROOM	8.93e-04
DMACHR----22W2LL	FAILURE OF COOLING COIL DMA-CC-22 IN EDG-2 ROOM	8.93e-04
SW-CHR-SCC1AW2LA	FAILURE OF SSW PUMP ROOM COOLING COIL PRA-CC-1A	8.73e-04
WMACHR---53AW2LL	FAILURE OF COOLING COIL WMA-CC-53A IN CRITICAL SWITCHGEAR ROOM	8.73e-04
DGACHR-DMA10W2LA	FAILURE OF COOLING COIL DMA-CC-10 IN EDG-1 ROOM	8.48e-04
DGACHR-DMA11W2LA	FAILURE OF COOLING COIL DMA-CC-11 IN EDG-1 ROOM	8.48e-04
SW-V-MODV-2BP2LB	MECHANICAL FAILURE OF MOTOR ACTUATED VALVE SSW-V-2B	8.47e-04
SW-V-MOV-12BP2LB	MECHANICAL FAILURE OF MOTOR ACTUATED VALVE SSW-V-12B	8.47e-04
RHRV-MO--48BN2LL	MECHANICAL FAILURE OF RHR-V-48B TO CLOSE ON DEMAND	8.24e-04
RHRV-MO--48AN2LL	MECHANICAL FAILURE OF RHR-V-48A TO CLOSE ON DEMAND	8.05e-04
SW-V-MODV-2AP2LA	MECHANICAL FAILURE OF MOTOR ACTUATED VALVE SW-V-2A	8.05e-04
SW-V-MOV-12AP2LA	MECHANICAL FAILURE OF MOTOR ACTUATED VALVE SW-V-12A	8.05e-04
EACENG-EDG-1U3D1	DIVISION 1 DIESEL GENERATOR EDG-1 IS UNAVAILABLE DUE TO CORRECTIVE MAINTENANCE	7.93e-04

FUSSELL-VESELY IMPORTANCE MEASURE		
SWATM	UNAVAILABILITY FROM SCHEDULED TEST/ MAINTENANCE OF SSW TRAIN A	6.39e-04
SW-PT--PS-1BW2LB	FAILURE OF PRESSURE SENSOR SSW-PS-1B SIGNAL	6.37e-04
EACENG-EDG-2U3D2	DIVISION 2 DIESEL GENERATOR EDG-2 IS UNAVAILABLE DUE TO CORRECTIVE MAINTENANCE	6.11e-04
SW-PT--PS-1AW2LA	FAILURE OF PRESSURE SENSOR SW-PS-1A SIGNAL	6.06e-04
EACENG-EDG-2T3D2	DIVISION 2 DIESEL GENERATOR EDG-2 IS UNAVAILABLE DUE TO TESTING	4.30e-04
EACENG-EDG-1T3D1	DIVISION 1 DIESEL GENERATOR EDG-1 IS UNAVAILABLE DUE TO TESTING	4.08e-04

Table 3.5-6

HUMAN ACTION FAILURE RATES

Basic Event ID	IPE Submittal Failure Rate per demand	Seismic PSA Failure Rate per demand	Description
ADSHUMNSTARTH3LL	2.66E-3	1.00E-2	Operator does not initiate ADS
HPSHUMNSTARTH3LL	2.66E-3	1.00E-2	Operator fails to initiate HPCS when required
LPSHUMNINITH3LL	2.66E-3	1.00E-2	Operator neglects to start LPCS when it is needed
RCIHUMNSTARTH3LL	2.66E-3	1.00E-2	Operator fails to initiate RCIC when required
RHRHUMNLPCISTART	2.66E-3	1.00E-2	Operator fails to initiate LPCI manually
RRAHUMNRFC10H3D2	2.66E-3	1.00E-2	Control room operator does not turn on RRA-FC-10 fan coil unit when required
RRAHUMNRFC11H3D1	2.66E-3	1.00E-2	Control room operator does not turn on RRA-FC-11 fan coil unit when required
RHRHUMNSP-COOLL	1.0E-5	1.00E-3	Human errors in following procedure PPM 2.4.2 to bring the RHR system into suppression pool cooling mode
RCIHUMNRCOLH3LL	0.002	0.025	Human error following procedure PPM 5.6.1 to provide alternate room cooling to RCIC

3.6 References

3.6.1 Generic Implementation Procedure, Revision 2

3.6.2 Electric Power Research Institute, "A Methodology for the Assessment of Nuclear Power Plant Seismic Margin," EPRI-NP-6041-SL, August 1991, Revision 1

3.6.3 Geomatrix, "Probabilistic Seismic Hazard Analysis WNP-2 Nuclear Power Plant Hanford Washington," Project No. 1846, December 1994

3.6.4 Electric Power Research Institute, "Methodology for Developing Seismic Fragilities," EPRI-TR-103059, April 1994

3.6.5 EQE, International, "Development of Seismic Fragilities for WNP-2 IPEEE," August 1994

3.6.6 EQE, International, "WNP-2 Seismic/Fire Interaction Assessment," March 1994

3.6.7 EQE, International, "Walkdown of WNP-2 for IPEEE," Calculation No. 59037-C-048

3.6.8 EQE, International, "IPEEE Building Forces," Calculation No. 59037-C-037, Revision 0

3.6.9 Supply System, "Motor Control Center Fragility Calculations," CE-02-95-04, February 1995

3.6.10 Supply System, "WNP-2 Review Level Earthquake (RLE) Relay Component Seismic Analysis for IPEEE," TM-2061, July 1995

3.6.11 Senior Seismic Review and Advisory Panel (SSRAP), "Use of Seismic and Testing Experience to Show Ruggedness of Equipment in Nuclear Power Plants," February 1991, Revision 4

3.6.12 EQE, International, "Screening Calculations for Equipment," Calculation No. 59037-C-040, Revision 0

3.6.13 EQE, International, "Probabilistic Spectra," Calculation No. 59037-C-048

3.6.14 Geomatrix, "Assessments of Dynamic Soil Properties for Soil-Structure Interaction Analyses and Soil Liquefaction Potential, Washington Public Power Supply System Nuclear Project No. 2," Project No. 2362, August 1994

3.6.15 Supply System, "Effect on Diesel Generators from Puncture or Catastrophic Failure of Nitrogen Tank CN-TK-1," ME-02-92-38, Revision 0, April 1993

3.6.16 Electric Power Research Institute, "Analysis of High Frequency Seismic Effects," EPRI-TR-102470, October 1993.

4.0 INTERNAL FIRE ANALYSIS

As part of the WNP-2 internal fire analysis, the plant's PSA fault tree and event tree models were used in conjunction with estimates of the frequency of occurrence of internal fires to arrive at core damage frequencies from specific fire areas in the plant. This procedure involved six individual sub-tasks:

1. selection of fire areas,
2. estimation of fire frequencies,
3. modelling of damage caused by a select group of hypothetical fires through simulation,
4. generation of conditional core damage probabilities given the occurrence of a fire,
5. determination of the likelihood for fire suppression, and
6. combination of fire initiation frequencies, fire suppression failure probabilities, and conditional core damage probabilities to arrive at core damage frequencies associated with internal fires.

The initial task, selection of fire areas, established compartments in the plant with which to determine the consequences of fires occurring within them, and to model the resulting plant response through the use of fault trees and event trees. The compartments are assumed to be independent from the others in the event of a fire occurring in any one of them. For the purposes of consistency, these compartments, or fire zones, are to a large extent those defined by the WNP-2 Fire Hazard Analysis (FHA) (Reference 4.10.4) developed in the previous WNP-2 Appendix R fire analysis. In a select number of instances, however, as discussed below, sub-compartments were established from those set down in the fire protection evaluation. These sub-compartments were used to reduce excessive conservatism where judgement, previous experience and/or fire damage simulations deemed them appropriate.

Next, fire frequencies were estimated for WNP-2 fire zones by the use of the FIVE ((Reference4.10.2) methodology. This NRC approved methodology draws upon United States nuclear power plant operating experience, and allows the analyst to apply the data in a plant-specific manner.

Third, to assist in justifying certain assumptions made in establishing fire zone boundaries and fire suppression capabilities, several computerized fire simulations were performed using the latest version of the fire simulation software program, COMPBRN IIIe (Reference 4.10.1).

Fourth, the generation of conditional core damage probabilities were prepared from modified forms of the WNP-2 fault trees and event trees. These modified models allowed the calculation of likelihood of core damage given the occurrence of a fire within each of the plant's fire zones. These modifications made it possible to account for the unavailability of plant systems and sub-systems assumed to be disabled by the occurrence of each fire. The unavailabilities of these systems and sub-systems were thus 1.0, and therefore when combined with the unavailabilities of other plant systems due to random causes (that is, other than those unavailabilities caused by fire damage), an adequate picture of the plant's ability to respond could be obtained.

Fifth, the likelihood for fire suppression was determined. This sub-task accounted for automatic initiation of fire suppression if available, as well as the likelihood that fires from specific combustion sources would not significantly effect the PSA-related components and cables located in the zone in question.

The sixth step in performing the internal fire accident sequence quantification involved the combining of internal fire initiation frequencies, fire suppression probabilities, and the conditional core damage probabilities to arrive at the final accident sequences for internal fires. This step was used to identify the important fire areas (defined as having calculated core damage frequency (CDF) greater than 1E-06/yr). For fire areas with calculated CDF > 1E-06/yr, a final step was conducted. This step included consideration of recovery actions for equipment assumed to fail, adjustment of initiation frequency based on location and positions of key cabinets and cable trays, and potentially limiting the amount of transient combustibles in some fire areas. Using these factors, the calculated CDF was recalculated.

The WNP-2 Control Room was analyzed separately as an individual fire area. The description and discussion of this analysis is contained in Section 4.6.6 in its entirety. The discussion in the following sections pertain to the remaining fire areas exclusive of the control room fire.

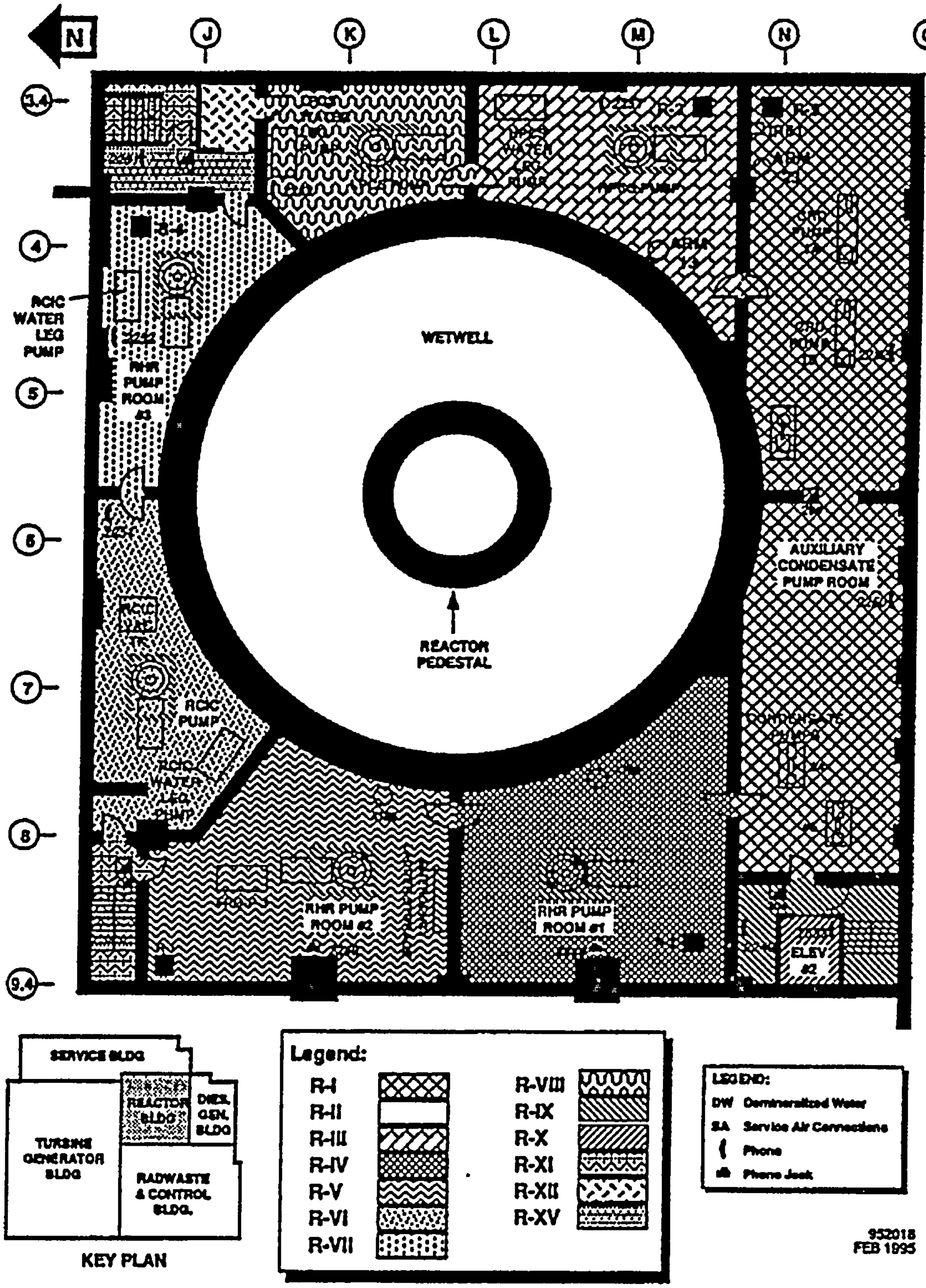
4.1 Selection of Fire Areas

The selection of zones used to perform the internal fire accident sequence quantification was based on those zones delineated in the WNP-2 Fire Hazards Analysis (FHA), in Appendix F of the WNP-2 FSAR. In certain instances, however, the WNP-2 fire zones were subdivided with the intent of removing excessive conservatism present in the FHA. The WNP-2 layout drawings provided in Figures 4.1-1 through 4.1-13 show the divisions graphically. In columns 1 and 2 of Table 4.2-1, a complete list of fire zones used in this analysis is provided.

Subdivision occurred for areas TG-1, and R-1, which are the Turbine Generator Building and Reactor Building, respectively. The criteria for subdivision of the fire areas is as follows. First, subdivision was assumed to be valid for areas separated by individual floors in the plant. For TG-1 and R-1, this meant that individual elevations were subdivided (e.g., R-1 was divided into elevations 441′, 471′, 501′, 522′, 548′, and 572′). Second, subdivision was assumed to be valid if judgement based on evaluation of barriers or distance or COMPBRN modelling warranted it. For TG-1, this meant that the corridor located on elevation 441′ containing a large percentage of the plant's safety-related cables was considered an independent area (named area TG-1K in Table 4.2-1). Also for TG-1, elevations 441′, 471′, and 501′ were each subdivided into three zones which corresponded to areas separated by distance and intermediate barriers such as walls or equipment (named areas TG-1A through TG-1H). In R-1, elevations 471′ and 501′ were each subdivided into four quadrants (named R-1B through R-1I) which were demonstrated to be independent in COMPBRN simulations as discussed in Section 4.4 of this analysis. Third, a number of equipment hatches and open stairwells exist in the plant. These were addressed by treating each as an individual fire area (named areas R-1M, R-1N, R-1O, and TG-1J) consisting of all equipment, combustion sources and ignition sources located within a twenty-foot perimeter of the open stairwell or equipment hatch.

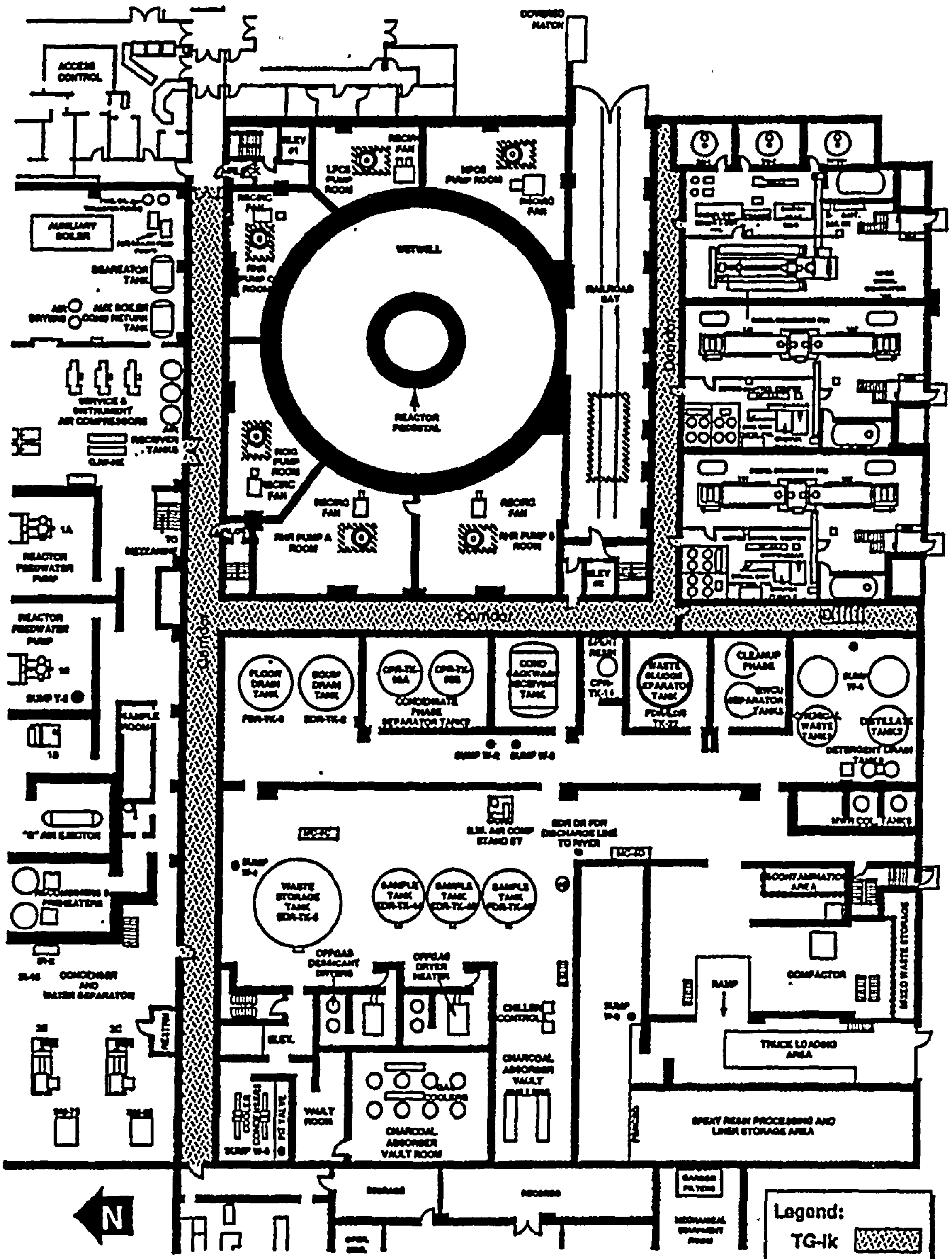
A list of all fire zones is provided in the first two columns of Table 4.2-1 and includes the zones that were subdivided. As can be seen from Table 4.2-1, area TG-1 was subdivided into 11 zones and area R-1 was subdivided into 15 zones.

As a result of this sub-task, 93 fire zones were established for WNP-2 for the purposes of this analysis and are listed in columns 1 and 2 of Table 4.2-1. Of these, 61 zones are identical to those established in the WNP-2 Fire Hazards Analysis. The remaining 32 zones comprise those formed from subdividing the Reactor Building (area RB-1 in the FHA), and the Turbine Generator Building (area TG-1 in the FHA) and those representing open equipment hatches and stairwells.



REACTOR BUILDING ELEV. 422'

Figure 4.1-1
FPSA Fire Areas, 422 ft. Elevation



8502028
April 1995

441' ELEV. TURBINE GEN. BLDG. CORRIDOR

Figure 4.1-2
FPSA Fire Areas, 441 ft. Elevation

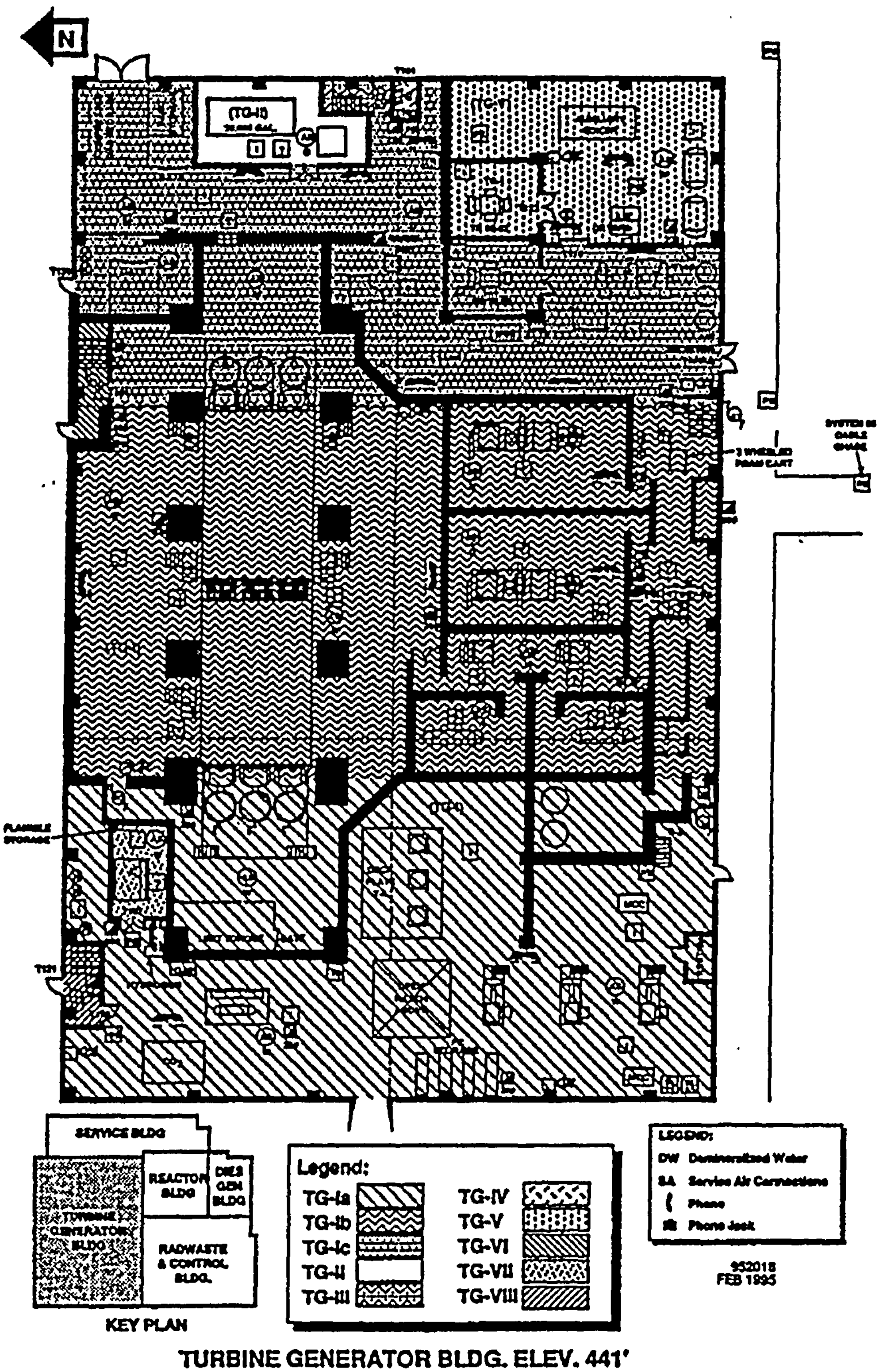


Figure 4.1-3
FPSA Fire Areas, 441 ft. Elevation

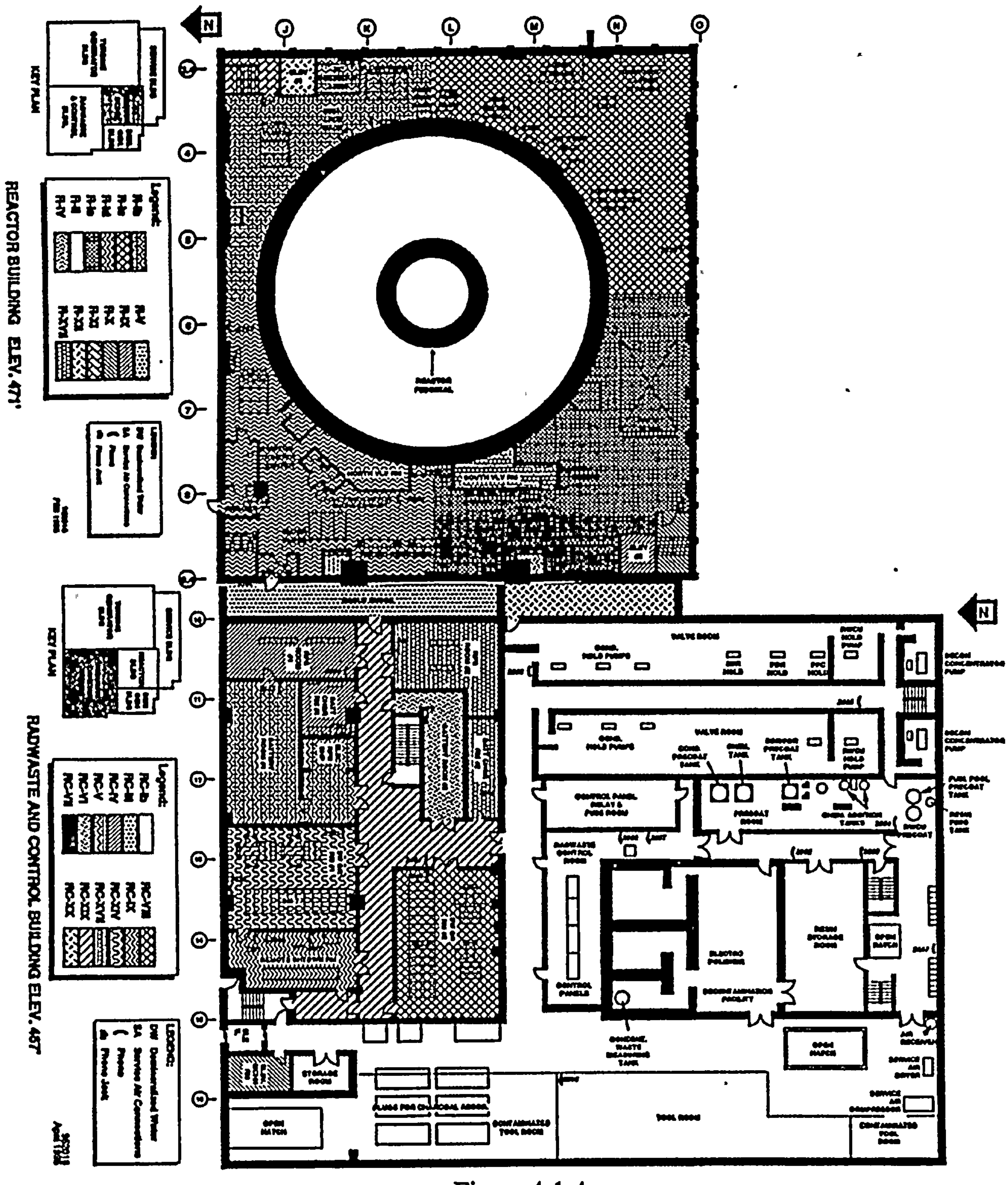
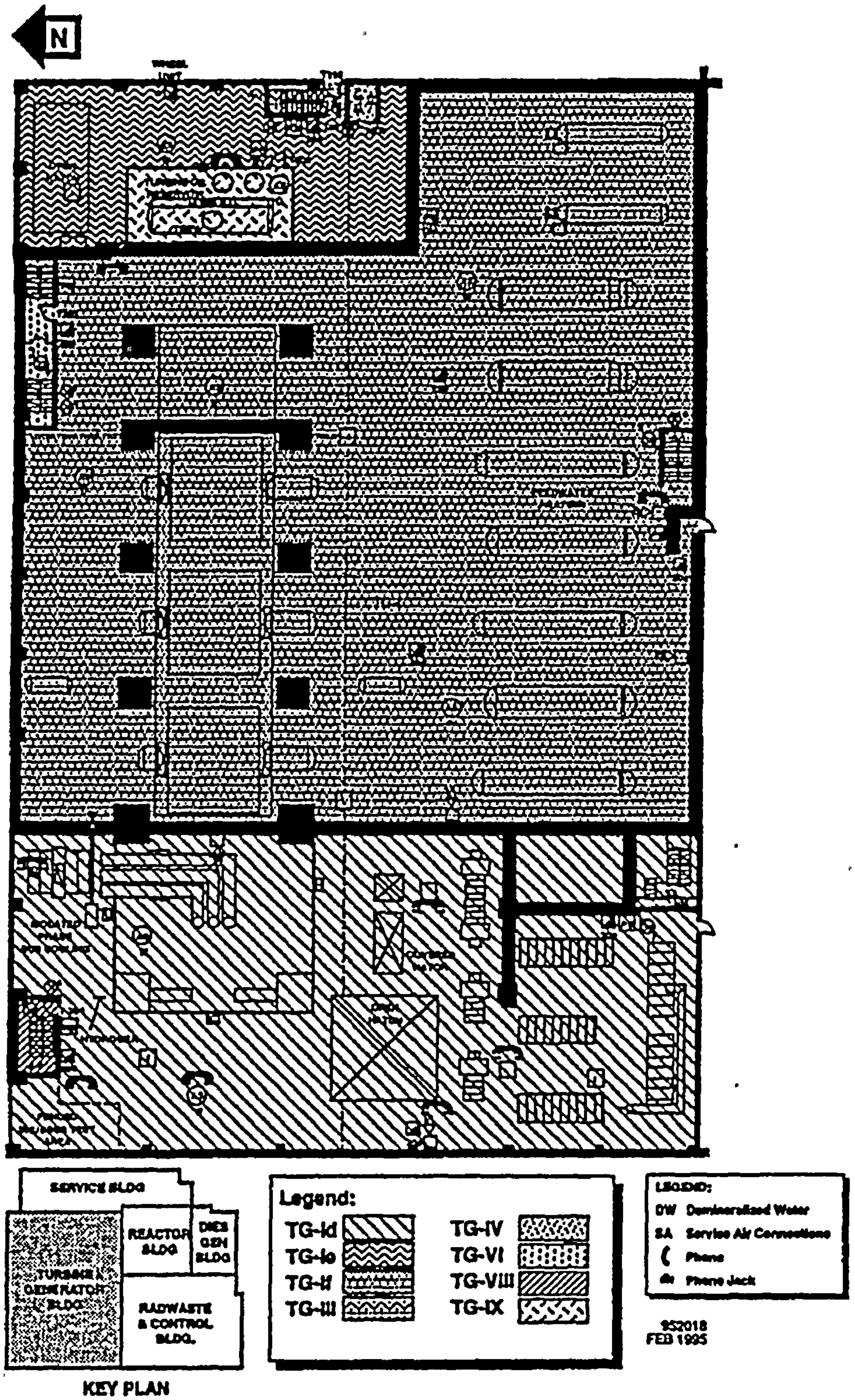
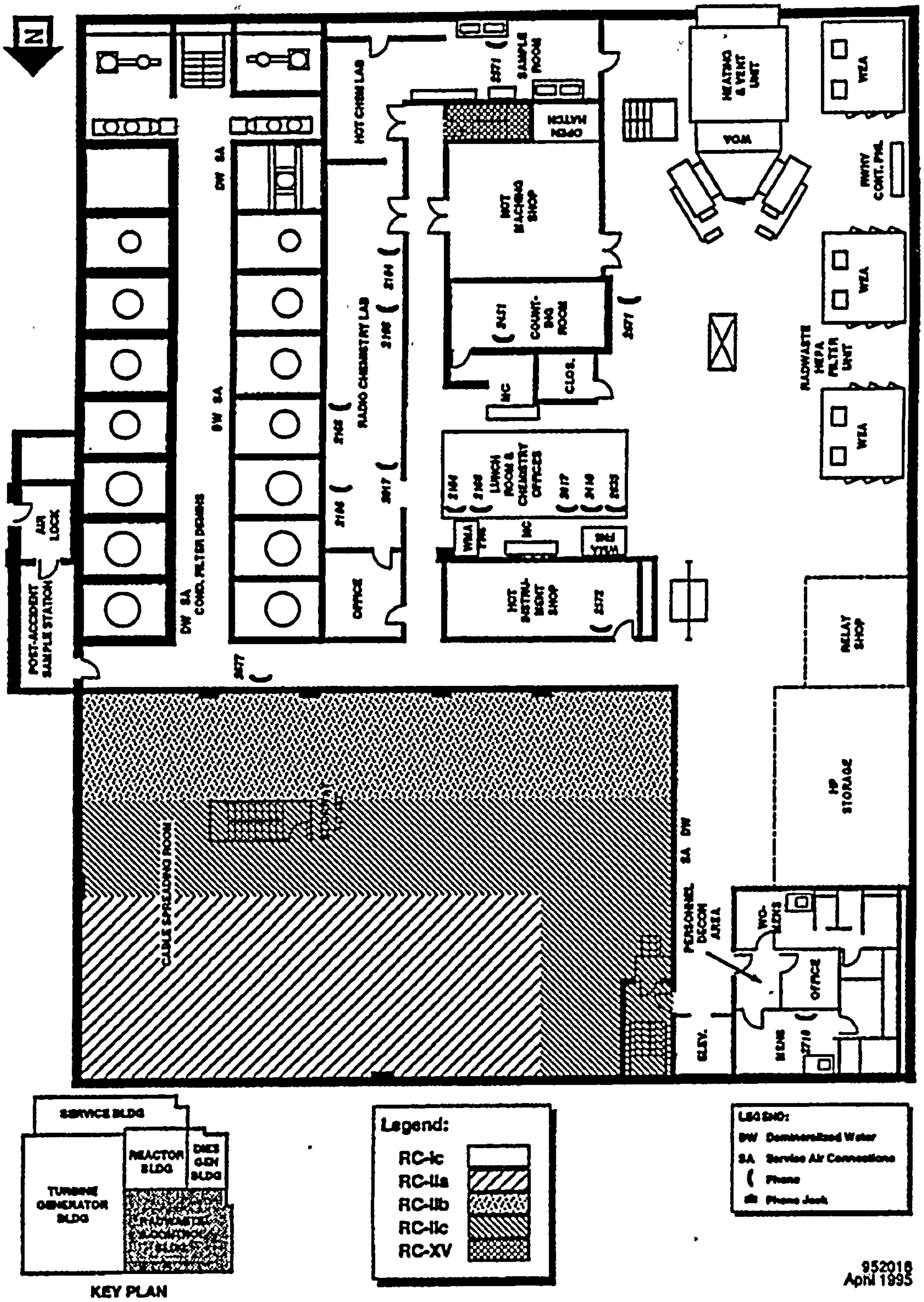


Figure 4.1-4
FPSA Fire Areas, 471 ft. Elevation



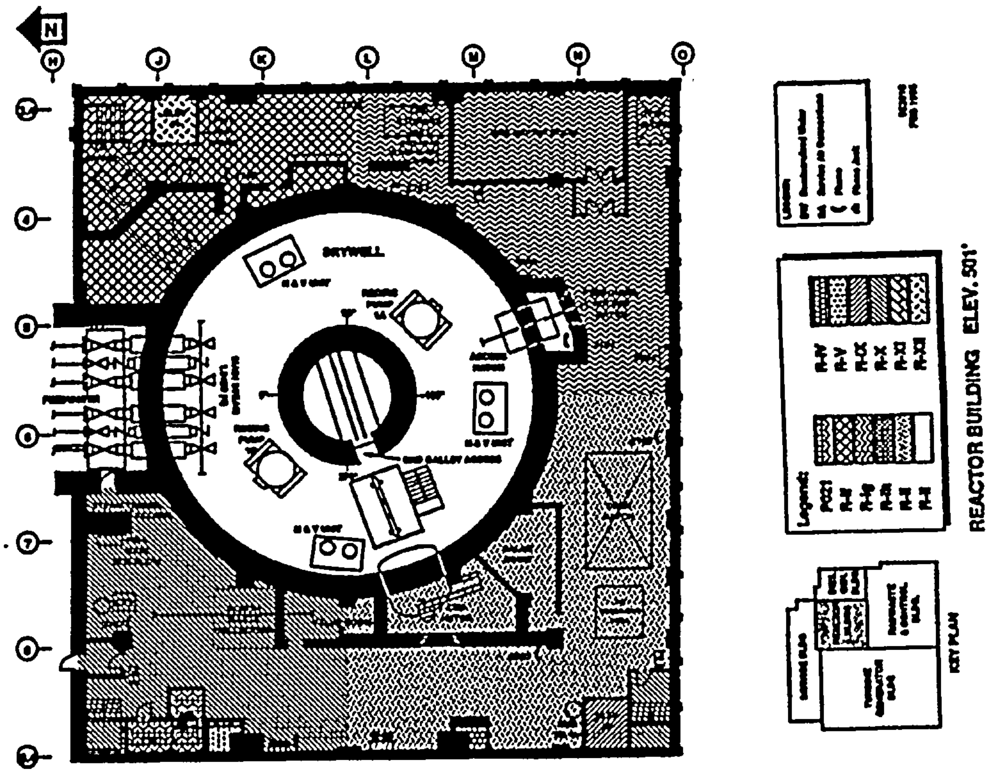
TURBINE GENERATOR BLDG. ELEV. 471'

Figure 4.1-5
FPSA Fire Areas, 471 ft. Elevation



RADWASTE BUILDING. ELEV. 487'

Figure 4.1-6
FPSA Fire Areas, 484 ft. Elevation



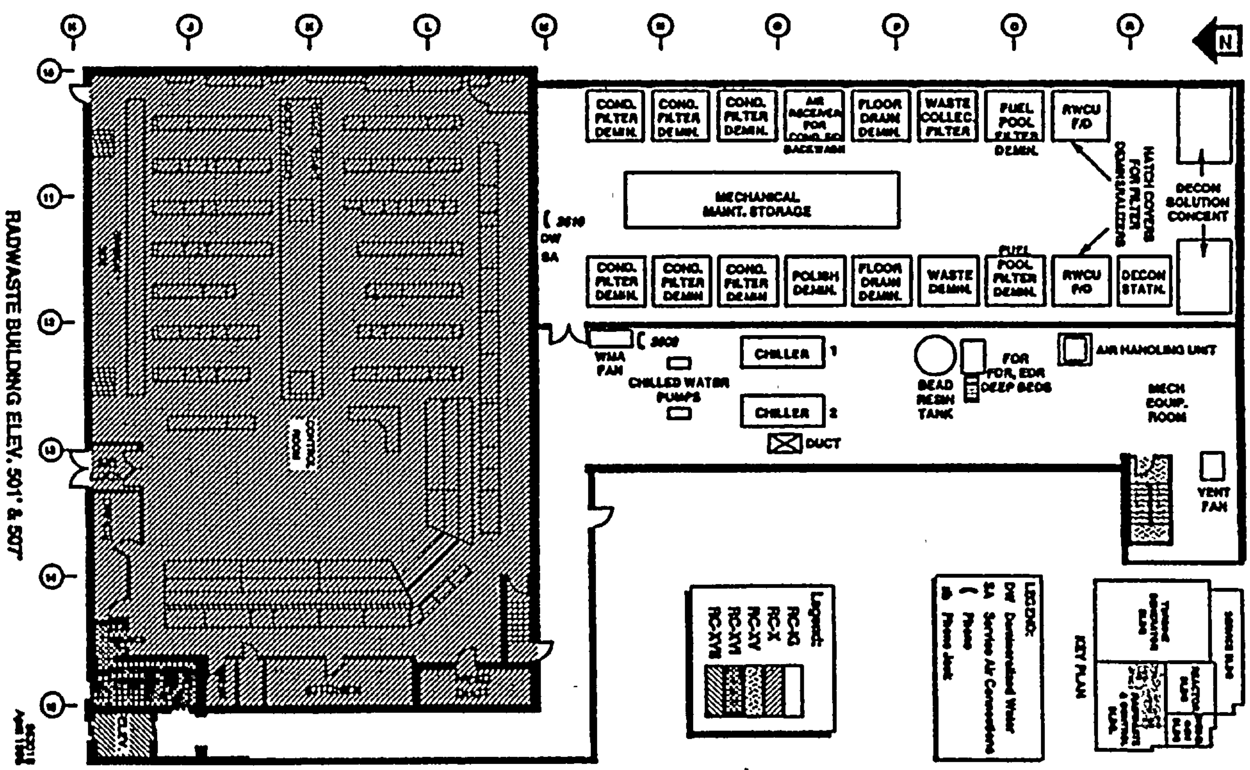
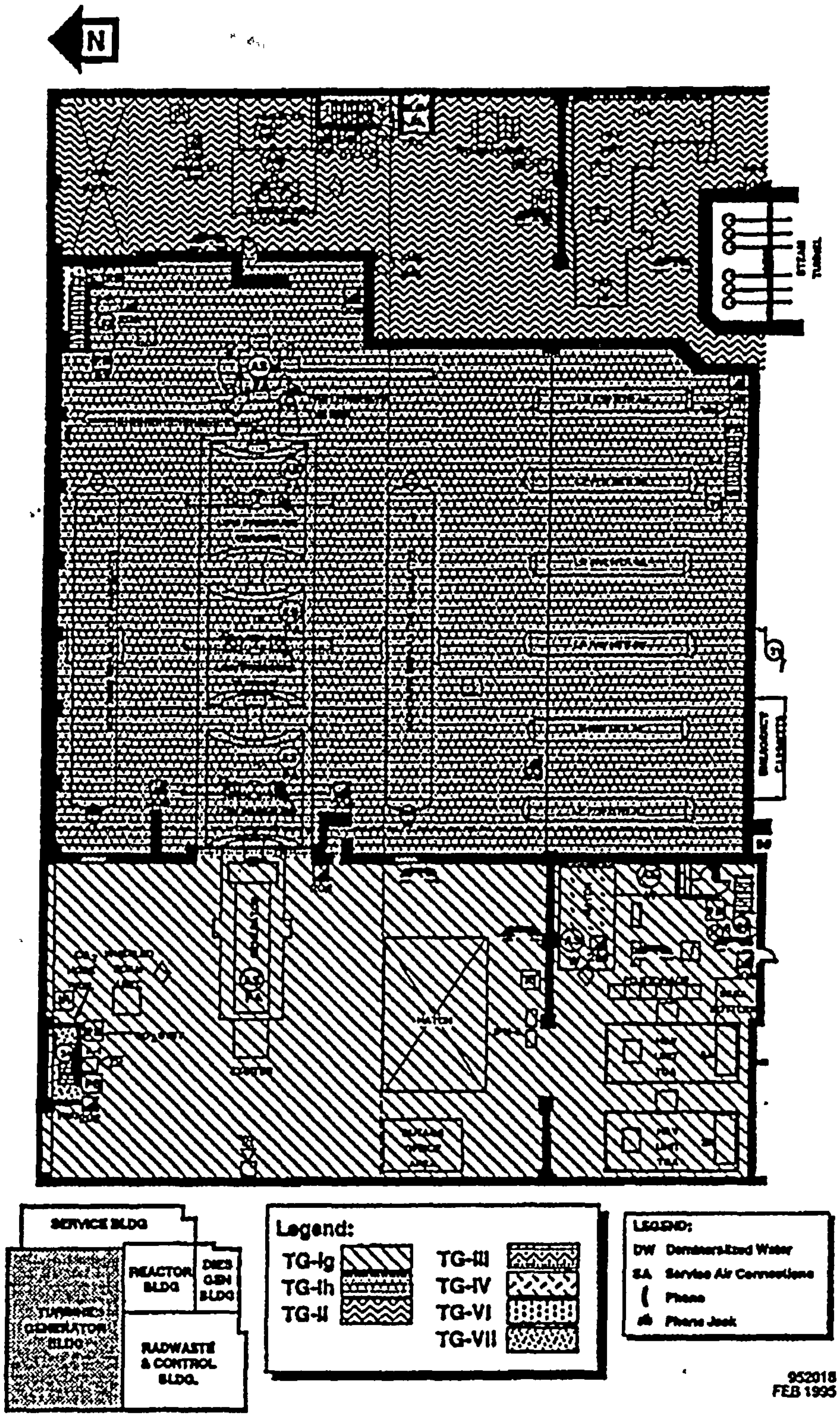


Figure 4.1-7
FPSA Fire Areas, 501 ft. Elevation



KEY PLAN

TURBINE GENERATOR BLDG. ELEV. 501'

Figure 4.1-8
FPSA Fire Areas, 501 ft. Elevation

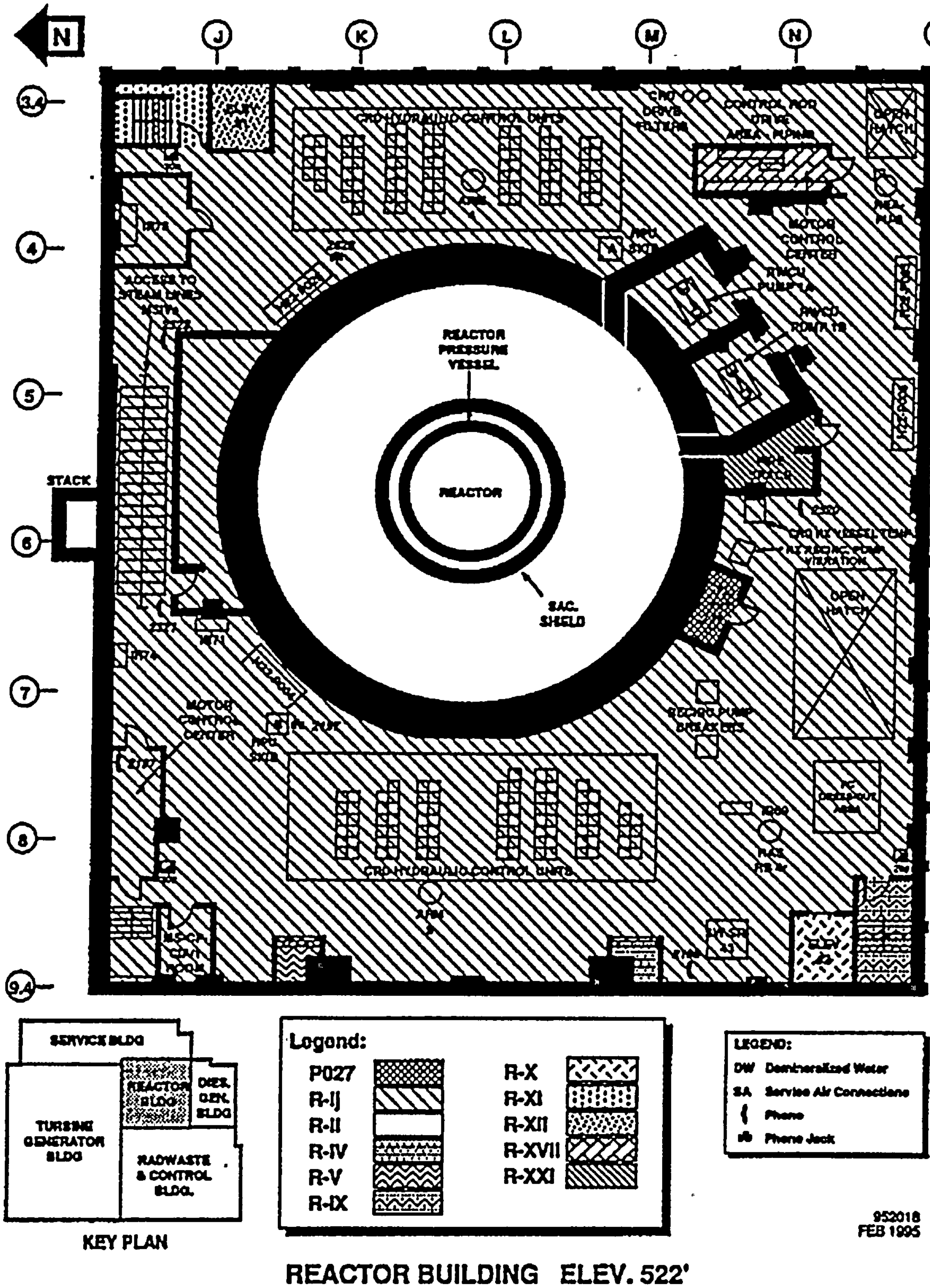


Figure 4.1-9
FPSA Fire Areas, 522 ft. Elevation

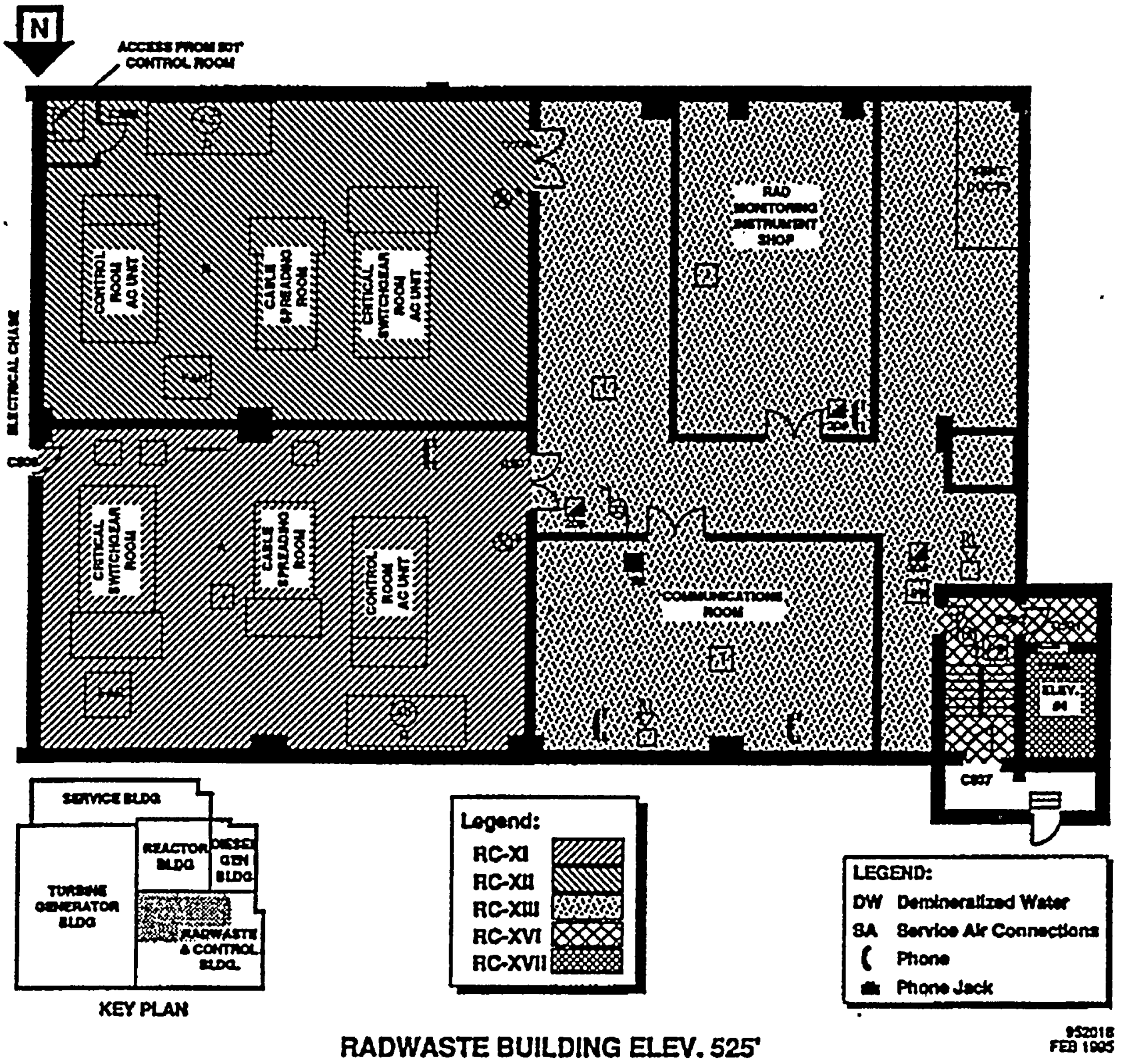


Figure 4.1-10
FPSA Fire Areas, 525 ft. Elevation

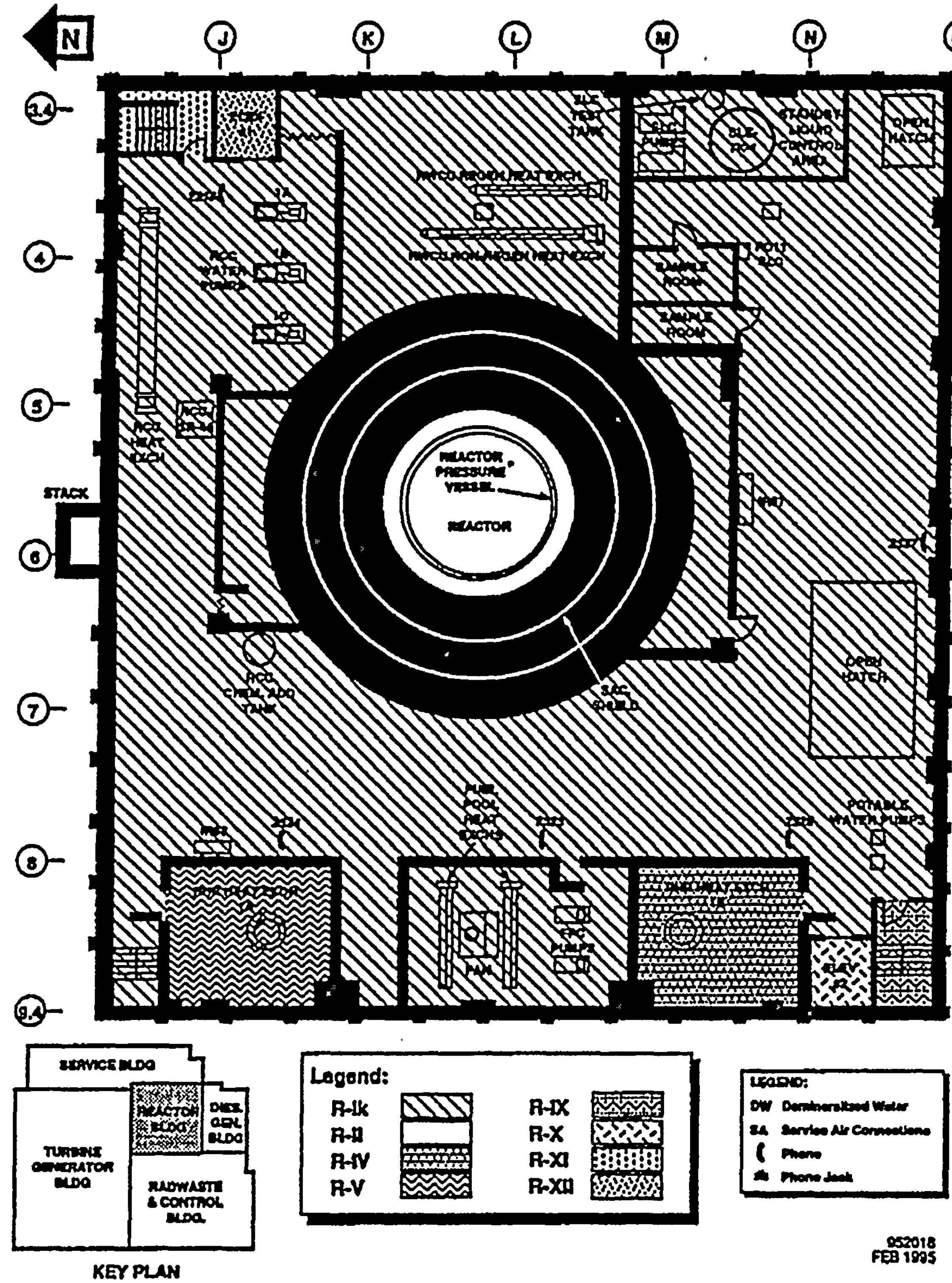


Figure 4.1-11
FPSA Fire Areas, 548 ft. Elevation

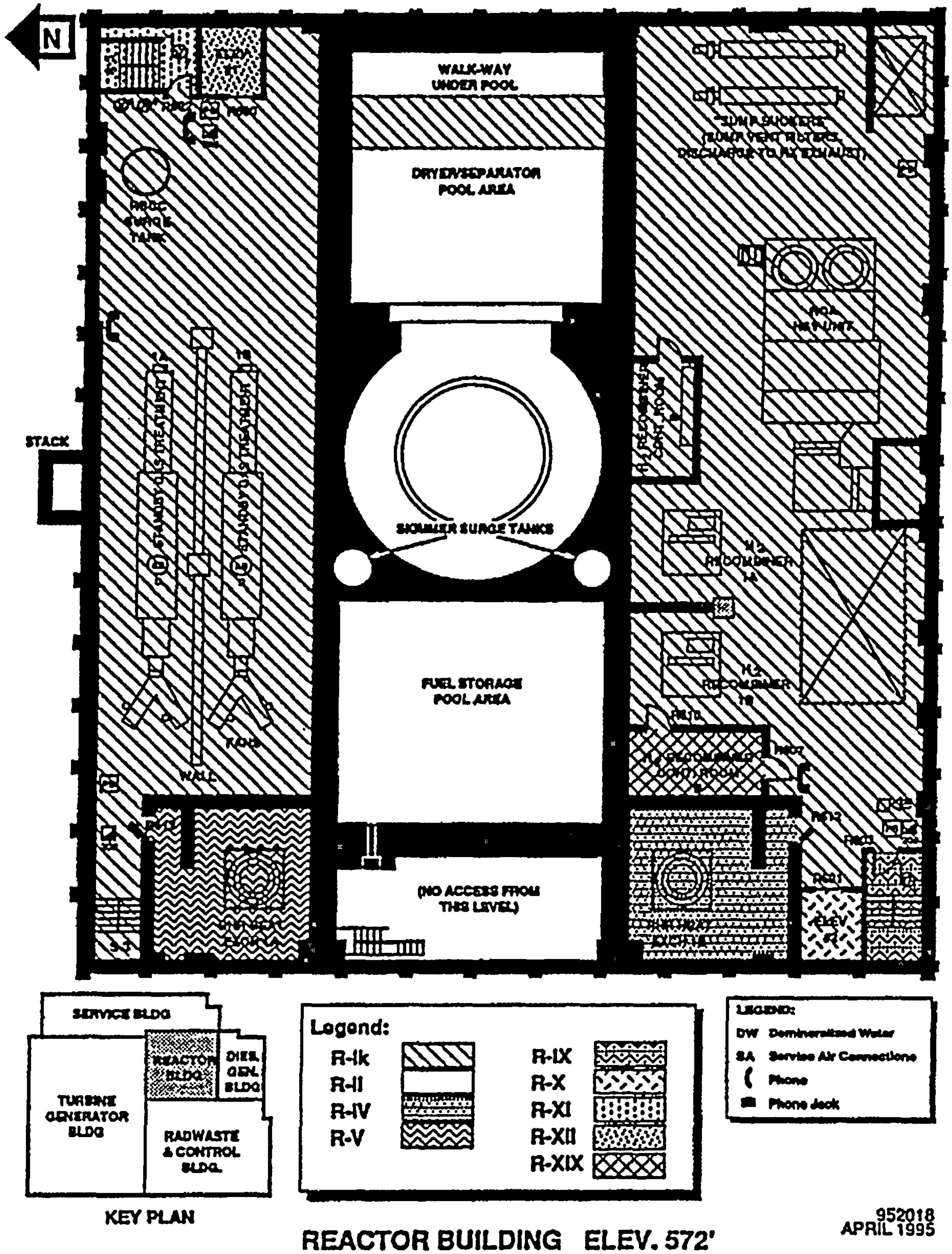


Figure 4.1-12
FPSA Fire Areas, 572 ft. Elevation

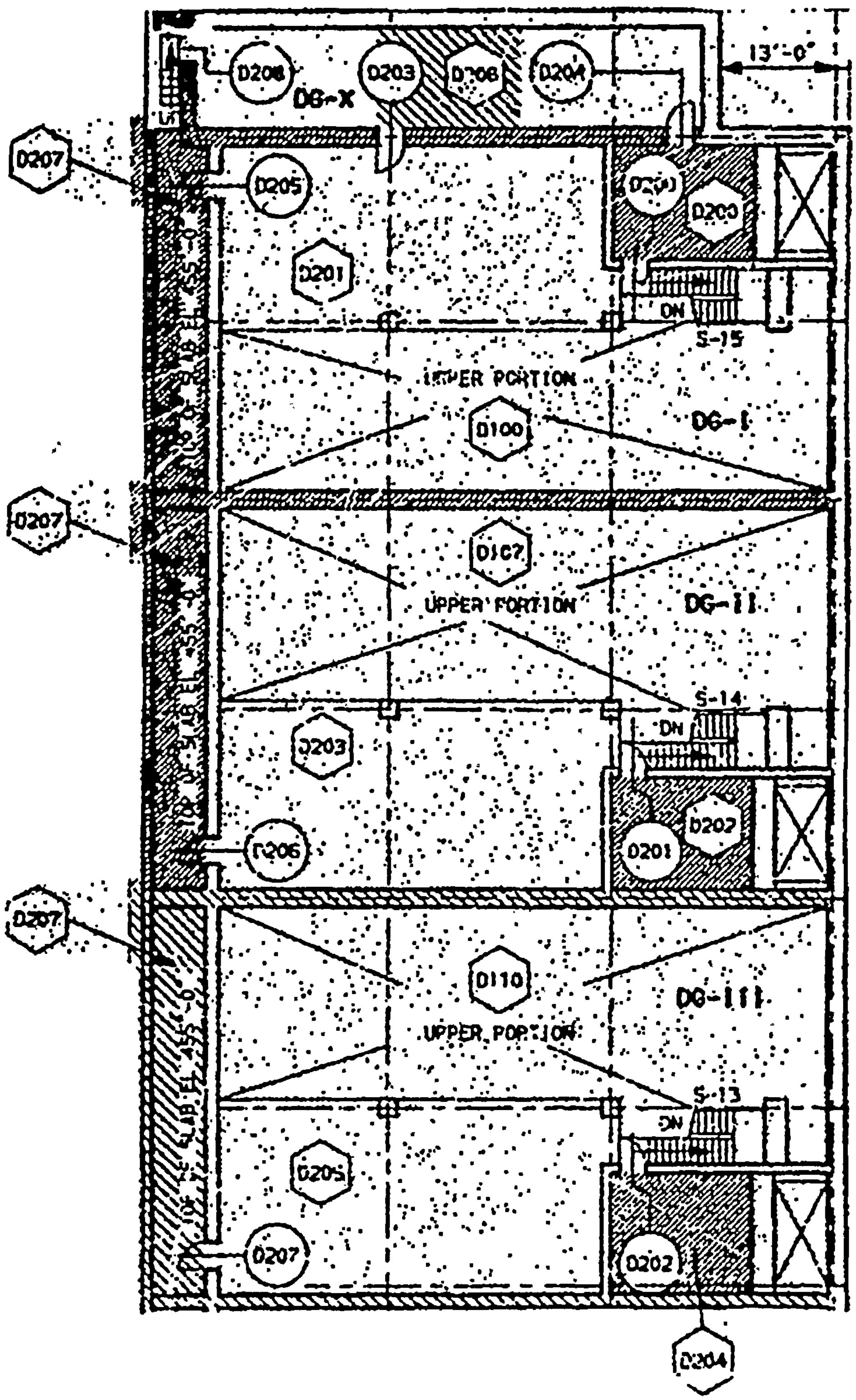


Figure 4.1-13
FPSA Fire Areas, 455 ft. Elevation

4.2 Fire Ignition Frequency Analysis

Fire frequencies were estimated for applicable WNP-2 fire zones as directed by EPRIs FIVE methodology for those areas which contain safe shutdown components or for which a fire would cause a demand for safe shutdown equipment.

The methodology employed to estimate fire initiation frequencies is provided in the FIVE document (Reference 4.10.2). The procedure involved the completion of one ignition source data sheet (ISDS) for each fire area under consideration as directed in the FIVE Methodology document. The ISDS sheets were prepared using information gathered from the Internal Fire Walkdown.

Using the FIVE Methodology guidelines as provided in Table 1.1 of the FIVE Methodology, the plant location weighting factors (WF_L) values are assumed to be as follows:

Reactor Building = 1/1 = 1
Diesel Generator Room = 3/3 = 1
Switchgear Room = 1/4 = 0.25
Battery Room = 1/2 = 0.5
Cable Spreading Room = 1/1 = 1
Intake Structure = 1/2 = 0.5
Turbine Building = 1/1
Radwaste Area = 1/23 = 0.044
Plant-Wide Components = 1

The FIVE Methodology prescribes the use of one of seven methods for calculating fire ignition source weighting factors, WF_{LS}, for each ignition source identified. These weighting factors normalize the FIVE generic data to reflect the plant-specific room configuration at WNP-2. The seven methods are lettered A through F and are delineated in Table 1.2 of the FIVE Methodology. The most appropriate weighting factor method was used for each ignition source in each area.

The fire ignition frequencies were obtained by taking the product of the plant location weighting factor, WF_L, the ignition source weighting factor, WF_{LS}, and the generic ignition source fire frequency, F_f. The frequencies incorporate consideration of both fixed as well as transient combustibles. In cases where the ignition source was qualified cable, and only in areas of totally qualified cable (cable spreading room, portion of turbine building corridor, and cable chase areas), an additional factor was included in the product to account for the ability of qualified cable to self extinguish. A factor of 0.05 was entered in these instances, which translates to a 95 percent probability that the cable will self extinguish. (This factor does not apply to those areas having automatically actuated fire suppression equipment. Detection and suppression are included later in the analysis and are discussed in Section 4.5.) The 0.05 factor was chosen on the basis that the fires involve qualified cable which propagates fire at a limited rate without additional external fire stimulation. There is, therefore, a certain likelihood that the fire would either self extinguish or would be detected and manually extinguished prior to the occurrence of damage to a significant quantity of cable. The importance of this assumption is analyzed in Section 4.9 and is certainly supported by studies such as contained in Reference 4.10.5.

The final results of the fire ignition source frequency calculations are shown in the last column of Table 4.2-1.

4.2-1 FIRE IGNITION FREQUENCY

PSA Fire Zone	Description	Zone Was Screened Away[1]	Fire Initiating Event Frequency[2]
DG-1	HPCS DG Room	N	3.1E-02
DG-10	Deluge Valve Rm	Y	NA
DG-2	DG Room #1	N	3.1E-02
DG-3	DG Room #2	N	3.1E-02
DG-4	DG-1 Oil Pump Rm	N	2.3E-03
DG-5	DG-2 Oil Pump Rm	N	2.3E-03
DG-6	HPCS DG Oil Pump	Y	NA
DG-7	HPCS Day Tnk Rm	Y	NA
DG-8	DG-1 Day Tank Rm	N	7.0E-04
DG-9	DG-2 Day Tank Rm	N	7.0E-04
R-10	Elevator	Y	NA
R-11	Stairway	Y	NA
R-12	Elevator	Y	NA
R-15	Lobby	Y	NA
R-17	S Valve Room	N	1.6E-03
R-18	Div 2 MCC Rm	N	2.9E-03
R-19	H2 Recombiner Rm	N	2.9E-03
R-1A	Railroad Bay 441′	Y	NA
R-1B	NE Reactor Bldg 471′	N	1.6E-03
R-1C	SE Reactor Bldg 471′	N	1.1E-02
R-1D	NW Reactor Bldg 471′	N	2.9E-03
R-1E	SW Reactor Bldg 471′	N	3.1E-03
R-1F	NE Reactor Bldg 501′	N	2.9E-03
R-1G	SE Reactor Bldg 501′	N	2.9E-03

4.2-1 FIRE IGNITION FREQUENCY

PSA Fire Zone	Description	Zone Was Screened Away[1]	Fire Initiating Event Frequency[2]
R-1H	NW Reactor Bldg 501′	N	2.3E-03
R-1I	SW Reactor Bldg 501′	N	7.9E-04
R-1J	Reactor Bldg 522′	N	8.2E-03
R-1K	Reactor Bldg 548′	N	1.4E-02
R-1L	Reactor Bldg 572′	N	1.4E-02
R-1M	Equipment Hatch	N	7.3E-03
R-1N	Stairwell (Uninstalled)	N	1.2E-03
R-1O	Stairwell S-3	N	1.6E-03
R-2	Drywell/Containment	Y	NA
R-21	S Valve Room	Y	NA
R-3	HPCS Room	N	5.0E-03
R-4	RHR-B Room	N	5.0E-03
R-5	RHR-A Room	N	5.0E-03
R-6	RCIC Room	N	2.8E-03
R-7	RHR-C Room	N	3.4E-03
R-8	LPCS Room	N	2.7E-03
R-9	Stairway	Y	NA
R-H22/P009	Instrument Rack Room	Y	NA
R-H22/P021	Instrument Rack Room	N	7.0E-04
R-H22/P027	Instrument Rack Room	N	7.0E-04
R-IR-73	Instrument Rack Room	Y	NA
RC-10	Main Control Rm	NA	See Section 4.6.6
RC-11	A A/C Room	Y	NA
RC-12	B A/C Room	Y	NA
RC-13	Emerg Chiller	Y	NA
RC-14	SWGR Rm #1	N	5.4E-03

4.2-1 FIRE IGNITION FREQUENCY

PSA Fire Zone	Description	Zone Was Screened Away[1]	Fire Initiating Event Frequency[2]
RC-15	Stairway	Y	NA
RC-16	Stairway	Y	NA
RC-17	Elevator	Y	NA
RC-18	Stairway	Y	NA
RC-19	Corridor	N	7.0E-04
RC-1A	Radwaste Bldg 437′	N	2.9E-03
RC-1B	Radwaste Bldg 467′	N	1.2E-03
RC-1C	Radwaste Bldg 487′	N	1.1E-02
RC-1D	Equipment Hatch	Y	NA
RC-1E	Equipment Hatch	N	1.1E-03
RC-1F	Equipment Hatch	Y	NA
RC-20	Pipe Chase	N	6.96E-04
RC-2A	Cable Spread Rm	N	7.46E-04
RC-2B	Cable Spread Rm	N	7.46E-04
RC-2C	Combust Free Zone	Y	NA
RC-3	Cable Chase	N	7.1E-04
RC-4	Div 1 Elec Equip Room	N	9.2E-03
RC-5	Battery Room 1	N	2.5E-03
RC-6	Battery Room 2	N	2.5E-03
RC-7	Div 2 Elect Equip	N	9.2E-03
RC-8	SWGR Rm #2	N	5.4E-03
RC-9	Remote Shtdn Rm	N	1.0E-02
SW-1	SSW Pump House 1A	N	4.0E-03
SW-2	SSW Pump House 1B	N	4.2E-03
TG-1A	Turbine Gen West 44	N	4.3E-03
TG-1B	Turbine Gen Center 4	N	6.3E-03

4.2-1 FIRE IGNITION FREQUENCY

PSA Fire Zone	Description	Zone Was Screened Away[1]	Fire Initiating Event Frequency[2]
TG-1C	Turbine Gen East 44	N	7.1E-03
TG-1D	Turbine Gen West 47	Y	NA
TG-1E	Turbine Gen East 47	Y	NA
TG-1F	Turbine Gen Center 4	N	7.0E-04
TG-1G	Turbine Gen West 50	Y	NA
TG-1H	Turbine Gen Center 5	Y	NA
TG-1I	Turbine Gen East 50	Y	NA
TG-1J	Equipment Hatch	N	4.1E-03
TG-1K	Turbine Gen Corr	N	7.11E-04
TG-2	Turbine Oil Storage	Y	NA
TG-3	Stairway	Y	NA
TG-4	Elevator	Y	NA
TG-5	Stair A3	Y	NA
TG-6	Stairway	Y	NA
TG-7	Hydrogen Seal Oil Rm	Y	NA
TG-8	Stairway	Y	NA
TG-9	Turbine Oil Reservoir	Y	NA

NOTES:

1 An entry of "Y" was made for zones that were screened from further analysis on the basis that the zone contained no PSA related components, no cables associated with these components, and no cables/components whose failure would cause a plant trip.

2 An entry of "NA" was made for zones screened away in the former column.

4.3 Review of Plant Information and Walkdown

In order to understand the effects a fire may have on any given fire area, it is necessary to understand what equipment may be damaged by a fire in any given area of the plant. All major equipment necessary for plant operation which, if lost, could contribute to core damage frequency is analyzed to determine where a fire may render the equipment inoperable. This list constitutes the fire PSA Safe Shutdown Equipment List (SSEL). For all equipment which may contribute to core damage frequency, a detailed circuit analysis is performed and the results entered into a data base. Cables associated with each component along with their associated cable routing were entered into the data base. The data base was then used to determine equipment losses for each fire area.

The data base was compiled using three equipment lists as sources; the level 1 WNP-2 IPE (Reference 4.10.3) list of basic events, the Appendix R safe shutdown equipment list and the equipment required for containment isolation (Reference 4.10.4).

First, the Appendix R Safe Shutdown Components were added to the database. Equipment EPN, cable numbers, cable routes, fire protection information, division and an 'R' in the type field were entered for each safe shutdown component. Cables which cause spurious actuation in the Appendix R analysis were so designated. The information entered into the database is a direct correlation of information from the Appendix R Safe Shutdown Analysis. No new circuit analysis or cable routing was completed for these components. Spurious cables which did not show cable routing were routed using CARPS (Reference 4.10.6), the WNP-2 controlled cable routing database, and entered into the database.

After addition of the Appendix R data into the database, the WNP-2 IPE basic event list was reviewed to determine components which may have a contribution to core damage frequency if damaged by fire. Equipment pieces which were a subset of larger components were not selected for further analysis, as these sub-components would be located during cable routing for the major component. The following items were not analyzed as discrete components. Cables associated with these components were, however, evaluated and included in the analysis.

- Relays
- Disconnects
- Switches
- Fuses
- Overloads

Manually operated components were also not selected based on non-susceptibility to fire. These components have no cables associated with them and therefore would not be affected by fire. Although access to these components could be affected by fire, such actions were not credited in the analysis in any way, and therefore, issues of accessibility do not apply. The following types of components were screened out by this method:

- Manual Valves
- Manual Check Valves
- Strainers
- Heat Exchangers

Basic events that were not associated with a component were screened out from further analysis. These basic events dealt with maintenance activities or operator actions. A fire would not cause these faults, therefore they were not evaluated.

All components which remained after review of the IPE basic event list were entered into the database. The selected components then had circuit analysis performed to determine which may be affected during a fire event. Current plant electrical drawings were used to select cables which would prevent operation of the component. In addition, cables which induce spurious operation were identified. The cables selected and associated information were transferred to Plant Equipment Cable Selection and Information Forms. Each cable selected was then routed using the CARPS database. The routing information is linked to compartments selected for the IPEEE Fire PSA. Each cable and fire area route was then entered into the fire IPEEE Master list database.

Upon completion of component selection and analysis from the IPE basic event list, the containment isolation fault trees were reviewed to determine if any components were present that were not within the IPE basic event list. The components present in the containment fault trees but not the IPE basic event list were also selected for inclusion in the fire SSEL and circuit analysis, and cable routing was performed on these components.

4.4 Fire Growth and Propagation

For all areas in the plant, a fire was assumed to destroy all equipment and cables in the area. Also, fire was assumed to be contained within a fire area on the basis that fire rated doors, dampers, penetration seals and concrete barriers will limit fire propagation. Therefore, the likelihood that fires propagate to more than one area is sufficiently low so as to not have a significant impact on the results. Equipment outside the area in which the fire ignited was assumed to be unaffected by direct fire damage. For some areas, this assumption was questionable and was investigated further with computer modeling of the fire growth and propagation.

In order to estimate the extent of damage which could potentially be caused by fires in selected areas of the plant, several fire damage simulations were performed using the COMPBRN fire modelling program. The methodology for utilizing the COMPBRN IIIe code is described in its User's Manual (Reference 4.10.1). Simulations were performed to find threshold levels of combustibles which could cause a given level of damage within selected fire zones. The methodology used for each simulation is discussed individually below along with the results of each.

No hot gas layer formed for the simulations in the Reactor Building and the Turbine Generator Corridor. Given the large size of the Reactor Building and the numerous openings it contains, the COMPBRN results showing that no hot gas layer forms was assumed to be accurate for fires occurring in the Reactor Building. Full ventilation was therefore assumed in the Reactor Building since no mechanism was identified in which the fusible links in the ventilation system would open to close the fire dampers.

A simulation was performed for Fire Area R-1B to determine the maximum amount of combustible located in one quadrant on the reactor building 471′ elevation which would not affect components located in an adjacent quadrant. The cylindrical wall surrounding containment was approximated by a straight wall in the simulation. For the purposes of modelling, room openings were represented by one opening with dimensions of 3 meters by 9.1 meters (the height of the zone) as only one opening can be designated in the program. The total area of all openings in the zone was estimated to be about 27 square meters based on scale plant layout drawings (Reference 4.10.4). The combustion source was representative of the combustion of two 55-gallon drums of oil burning with lids removed.

The physical property parameters assumed for objects modelled in the simulations were obtained from data supplied with COMPBRN IIIe with one exception. The physical property parameters related to cable damage were assumed to be the following (Reference 4.10.9):

Spontaneous Ignition Temperature = 932°F
Damage Temperature = 662°F

As would be expected, the cable trays located directly over the two burning 55-gallon drums ignited and were damaged in the simulation. The cable trays located in the adjacent quadrant (the "Target" trays in the COMPBRN analysis), however were not damaged. Thus, the combustion of two 55-gallon drums has been demonstrated to be the threshold at which cables located in adjacent quadrants of the Reactor Building 471′ elevation remain unaffected.

WNP-2 design basis combustible loading calculations conservatively assume that a single 55 gallon drum of oil will bound the transient combustibles in a given fire area.

A COMPBRN IIIe analysis was performed to support subdividing the 501′ elevation of the reactor building into four quadrants. The simulation for the Reactor Building 501′ elevation was essentially identical to that for elevation 471′ described above. The ceiling is not as high on elevation 501′, however, the same threshold of two 55-gallon drums was demonstrated to be applicable. The eight quadrants on elevations 471′ and 501′ of the Reactor Building (areas R-1B through R-1I in Table 4.2-1), are established to be justifiably independent fire zones based on these two analyses. Based on the simulations above, a fire postulated to occur in one quadrant will not significantly affect components located in adjacent quadrants.

COMPBRN IIIe analysis was performed for an electrical cabinet fire. The methodology employed for this simulation is similar to that performed for the Reactor Building 501′ elevation as described above. However, the combustible in this simulation was intended to represent an electrical cabinet fire. This simulation was performed by igniting 15 kg of cable insulation material located 2 meters above the floor (this conservatively assumes that all electrical insulation in the cabinet is located at the cabinet's top) and igniting it by burning a small quantity of oil located adjacently. The simulation results indicate that the tray located over the cabinet is damaged at two minutes. Cables located more than 1.2 meters from directly overhead the cabinet were not damaged. This simulation indicates that an electrical cabinet fire may damage cables located directly over the cabinet, but that cables located elsewhere in the vicinity would not be damaged.

A COMPBRN IIIe analysis was performed for a fire in valving, lubricating oil in electric pumps and valves. This simulation was performed in a manner similar to that for the electrical cabinet fire described above except that lubricating oil associated with a pump is used as the combustible. Here, the quantity of oil was 10 kg (about 3 gallons) which is conservatively assumed to be representative of the quantity used to lubricate each pump and valve located in the reactor building (other than the ECCS pumps). None of the cable trays modelled in this simulation were damaged suggesting that fires caused by combustion sources of this type (pumps and valves) do not significantly affect safety-related components installed nearby in the Reactor Building.

COMPBRN IIIe simulation was also performed to determine the threshold quantities for various combustibles for which the cable spreading room 20-foot combustible free zone, RC-IIC, would be effective (that is, would not allow propagation of fire damage between zones RC-IIA and RC-IIB).

The simulation was performed by placing the combustible on the edge of RC-IIA located closest to RC-IIB. Several levels of cable trays were modeled as if they were in place over the combustible. Several more levels of cable were modeled in RC-IIB as "targets" to test the effectiveness of the combustible-free zone.

Ventilation dampers are designed to close in rooms containing safety-related equipment if the fusible-links installed within the ventilation ducts reach approximately 165 degrees F. For areas in which COMPBRN indicated that a hot gas layer formed, such as the cable spreading room, the 165 degree F temperature was exceeded in the first 60-second COMPBRN iteration. Therefore, for the COMPBRN runs for these areas, no ventilation was assumed since the simulations show that the 165 degree F threshold is exceeded in such a short time.

Three different simulations were performed for the Cable Spreading Room. The simulations demonstrated the combustible threshold for paper, paint, and solvent. The thresholds were found to be 60 pounds of paper, or four 5-gallon cans of paint, or four 5-gallon cans of solvent. Combustion of these materials in three individual simulations resulted in no damage occurring to cable trays located on the opposite side of the combustible-free zone. Thus, the cable spreading room combustible-free zone (RC-IIC) was found to be effective during fires involving these quantities of fuel.

The corridor between the reactor building, the radwaste building, and the turbine generator building, Zone TG-1K, was also modeled to establish the quantity of transient combustibles required to damage electrical cables in the area. This simulation was performed by assuming that the corridor is configured as one long, straight corridor to allow for COMPBRN modelling limitations. Since it was unclear from the simulation whether the corridor's ventilation dampers would close on high temperature (due to the fusible links), two simulations were performed: one assumed full ventilation, the second assumed no ventilation.

The combustible threshold was demonstrated by COMPBRN to be two 55-gallon drums of oil with lids removed for both ventilation simulations. No damage occurred to cables given this amount of combustible. The flame height was demonstrated to be very close to the level at which cables are installed in the corridor. However, given the high corridor ceilings and distance from the fire source (oil drums) to cable trays, no hot gas layer forms and the cable temperature does not reach damage temperature. It is judged that the results of these simulations showing that no damage occurs carry a significant degree of uncertainty. However, since the simulations incorporate a combustible quantity double of that actually allowed to be present in the area, it is judged that the simulations are adequate despite uncertainties.

4.5 Fire Detection and Suppression

Credit for fire suppression was taken for fire zones having automatic fire suppression systems installed. These areas are RC2A, RC2B, and DG1 through DG9. The unavailability for such suppression was assumed to be 0.025/demand (Reference 4.10.2) and this was used as the probability for failure to suppress. No other credit was taken for fire suppression.

4.6 Analysis of Plant Systems, Sequences and Plant Response

Two important tasks in a fire PSA involve identifying the equipment that is failed by cable damage caused by a fire in a given area and then relating that equipment unavailability to a conditional probability of core damage. These tasks require computerized sorting of the cable routing data base and then manipulation of the fault tree and event tree models in the IPE.

The IPE study contains fault tree and event tree models from which we can generate combinations of component failures which will lead to core damage. The baseline IPE calculates core damage frequency assuming the components fail randomly. It is possible to manipulate the fault tree models to postulate several components failing at the same time due to the same cause (such as a fire in one room). The core damage frequency can then be re-calculated with the undamaged components failing at their random rate.

For the first quantification, the fire PSA assumes all cables in a given fire zone or fire area are damaged. All components and equipment which require these cables to function are then inoperable. This fire study included the effects of failed power cables, control cables, and instrumentation cables. Power cables and control cables are usually essential to component operation, whereas instrumentation cables can often fail with no effect on the component. Appendix R circuit analysis was used to determine which instrumentation cables were essential for component operation.

This fire study started with the premise that the IPE includes all components necessary to prevent core damage (for initiators other than fires). The list of Appendix R components was reviewed to determine if any additional components need be included in the cable tracing effort. The cable routing information for all cables is contained in the CARPS data base at WNP-2. The CARPS information can be sorted by specified geographic location to provide cable resident information for any room. For each fire zone established by the PSA, the physical locations of the zone boundaries (i.e., floor height, ceiling height, and wall coordinates) were specified and the cables within those boundaries were identified. The cables are then related back to the components they serve to produce a list of components failed by a fire in a room. The components are then related to system or subsystem failure.

Conditional core damage probabilities were quantified using the WNP-2 fault trees and event trees as a basis to establish accident sequences that account for components assumed to be disabled by internal fires within the plant's fire zones, and random unavailabilities of components unaffected by the postulated fires.

To account for the complete unavailability of WNP-2 systems or trains due to internal fire, house events were inserted into the plant's existing fault trees. These events act like switches, and when their value is 1.0, the train or system to which they're connected is completely failed when the fault tree is solved.

The specific settings for the house events (either 1.0 (on) or 0.0 (off)) are summarized in the matrix shown in Table 4.6-1. The entries marked by X indicate that the corresponding component/system is failed, and the applicable house event was changed from a value of 0.0 to 1.0 (failed). The house events have labels that correspond to the entries made in the top row of the matrix (e.g., any failure of the SM-7 bus was noted with an X in the column marked SM7 and this meant that the house event labelled XHOS-SM7 was changed from a value of 0.0 to a value of 1.0).

Rather than produce an event tree for every unscreened fire area, the fire areas were grouped according to similar damage effects. To assist in the fire area grouping effort, dependent failures were indicated in Table 4.6-1. A "D" in the table indicates that the system itself was not affected, but that a required support system was disabled. After grouping, 34 event trees were required.

The event trees were solved using an initiating frequency of 1.0 and therefore represent conditional core damage frequencies. The conditional core damage frequencies were then combined manually with the ignition source frequency and fire detection and suppression frequency for each area. This allows a clear picture of the contribution of each fire area to the overall core damage frequency.

The applicable event trees used to quantify internal fire accident sequences were the loss of feedwater event tree (TF) and the turbine trip event tree (TT). LOCA events induced by hot shorts were considered to be addressed by the Appendix R analysis and sufficient means of responding to such situations have been incorporated into the plants control systems. Therefore, the frequency of hot short induced LOCA initiators when coupled with the likelihood that operators fail to respond, make such initiators insignificant contributors to the overall internal fire core damage frequency contribution. For LOSP events induced by hot shorts, the TF event tree was used and the loss of offsite power was accounted for in the system models. Hot shorts that result in inadvertent actuation of systems were considered to be adequately addressed by the Appendix R analysis and that such events when coupled with the likelihood for failure to respond are not significant relative to the scenarios addressed in this analysis. The TF event tree structure was used in any instance where the power conversion system was failed by loss of feedwater, loss of condensate, or loss of main steam system as a result of the fire. The TT event tree structure was used in all other instances. The branch structure of these trees was left unchanged for this analysis. The manner in which they were solved and the supporting fault tree solutions, however, were different in order to account for system failures caused by fire. When modeling certain fire events, the combined failure of components results in the complete failure of event tree functional headings. In such instances, the NUPRA Code provides a message stating that the top event is "True" and therefore cannot be solved. To deal with this situation, a single event equation is used which has a value of 1.0. Thus, the event tree structure for all fire events based on turbine trip have the same functional headings as the turbine trip event tree however, some branch points have been effectively removed by using a failure rate of 1.0.

Sequence Quantification Assumptions

1. Functional event V_4 in the internal fire event trees (SW Train B crosstie to RHR-B Available) was assumed to be unavailable for instances in which Service Water Train B was disabled by internal fire.

2. In instances where internal fire causes MSIV closure, recovery of the MSIVs was not credited for the initial screening calculation. Therefore, functional event Z in the internal fire event trees was assumed to fail with a probability of 1.0.

3. Offsite power was assumed to be available unless the fire damage indicated otherwise. If offsite power was unavailable, the power conversion system was assumed to be unrecoverable.

4. Loss of room cooling to a major electrical bus room was assumed to result in loss of the affected bus. In reality, opening doors and/or portable fans would prevent failure.

5. All equipment and non-fire protected cables were assumed to be completely failed for each zone evaluated in the initial screening analysis. For the refined analysis discussed in Section 4.6.4, some consideration was given for failures of subgroups of components.

TABLE 4.6-1: FIRE DAMAGE EQUIPMENT LIST

TG-1J	TG-1F	TG-1B	TG-1A	SW-2	SW-1	RC-20	RC-19	RC-14	RC-9	RC-8	RC-7	RC-6	RC-5	RC-4	RC-3	RC-2B	RC-2A	RC-1E	RC-1C	RC-1B	RC-1A	DG-9	DG-8	DG-5	DG-4	DG-3	DG-2	DG-1	
F3	F3	F3	F9	F25	F20	F18	F31	F8	F5	F14	F33	F34	F30	F32	F16	F10	F13	F1	F17	F3	F11	F25	F20	F22	F28	F26	F28	F29	FIRE MODEL EVENT TREE
X	X	X	X												D					X	X								COND
X	X	X	X										D		D					X	D					D			RFW
							D	D	D	D	D			D	D	D	D				X								PCS
							X	X	X	X	X			X	X	X	X				X								MSIV Closure
								X	X	X	D		D	X		X	X												VENT
													X		X														SM-7 (MC-7F, SL-73)
			X					X									X				X								230KV to SM7
						X		X					X		X		X								X		X		115KV to SM7
													X	X	X										X		X		MC-7A (PP-7A, MC-7AA, PP-7AE)
													X	X	X														MC-7B (MC-7BA, MC-7BB)
																													MC-7C (MC-7CA)
																			X										MC-7E (MC-7EA)
						X																					X		SM-75
											X																		SM-8 (SL-83, MC-8F)
			X						X	X						X	X				X			X					230KV to SM-8
			X						X	X						X								X		X			115KV to SM-8
											X				X											X			MC-8A (PP-8A, PP-8AA, PP-8AC)
																													MC-8B (MC-8BA, MC-8BB)
																													MC-8C (MC-8CA)
																													SM-85
			X														X				X								230KV to SH5
			X														X												230KV to SH6
					X			X					X	X	X		X						X		X		X		DG-1
				X					X	X	X					X						X		X		X			DG-2
															X					X	X							X	DG-3
							X			X	X	X			X	X										X			DP-S1/2 (DP-S1/2D, DP-S1/2E)
			X					X					X	X															DP-S1/1 (DP-S1/1C)
							X			X	X	X																	DP-S1/7
														X															DP-S2/1
													X																B1-1
													X	X															C1-1
													X																B2-1
													X																C2-1
												X																	B1-2
											X					X													C1-2
												X																	B1-7
										X	X																		C1-7
					X										X	X	X											X	HPCS
			D					X	X				D	X	X	X	X												RCIC
					X			X					X		X										X		X		SW-A
					D	X		X					X	X	X		X												RHR-A
					D	X		X					X	X	X		X												LPCI-A
				X					X	X	X					X								X					SW-B
				D					X	X	D					X								X					RHR-B
				D					X	X	D					X								X					LPCI-B
																													SW-C
									X	X					X	X	X											X	LPCI-C
			D		D			X					X	X	X	X	X								D		D		LPCS
															X			X			X								TSW-A
															X						X								TSW-B
														X	X		X		X		X								CAS-A
							X				X				X	X													CAS-B
															X		X												CAS-C

X - System or Train affected by Fire
D - Dependant Equipment

TABLE 4.6-1: FIRE DAMAGE EQUIPMENT LIST (Continued)

R-P27	R-P21	R-19	R-18	R-17	R-8	R-7	R-6	R-5	R-4	R-3	R-1oh	R-1N	R-1M	R-1L	R-1K	R-1J	R-1I	R-1H	R-1G	R-1F	R-1E	R-1D	R-1C	R-1B	TG-1K	TG-1C	
F27	F19	F19	F27	F19	F21	F21	F25	F18	F27	F29	NA	F29	F6	F23	F23	F6	F18	F6	F23	F24	F2	F7	F4	F7	F15	F12	FIRE MODEL EVENT TREE
																											COND
																										X	RFW
											D		D			D		D			D	D	D		X		PCS
X											X		X			X		X			X	X	X		X	X	MSIV Closure
																									X		VENT
																											SM-7 (MC-7F, SL-73)
																									X		230KV to SM7
																									X	X	115KV to SM7
																											MC-7A (PP-7A, MC-7AA, PP-7AE)
											X		X	X	X	X											MC-7B (MC-7BA, MC-7BB)
												X	X						X	X			X	X			MC-7C (MC-7CA)
																											MC-7E (MC-7EA)
																											SM-75
																											SM-8 (SL-83, MC-8F)
																									X		230KV to SM-8
																									X	X	115KV to SM-8
																											MC-8A (PP-8A, PP-8AA, PP-8AC)
		X	X																								MC-8B (MC-8BA, MC-8BB)
																											MC-8C (MC-8CA)
																											SM-85
																									X		230KV to SH5
																									X		230KV to SH6
																									X		DG-1
							X																				DG-2
										X	X	X										X	X	X			DG-3
																											DP-S1/2 (DP-S1/2D, DP-S1/2E)
																											DP-S1/1 (DP-S1/1C)
																											DP-S1/7
																											DP-S2/1
																											B1-1
																											C1-1
																											B2-1
																											C2-1
																											B1-2
																											C1-2
																											B1-7
																											C1-7
										X	X	X										X	X	X	X		HPCS
			X			X	X	X	X	X	X	X	X		X	X		X	X	X	X	X		X	X		RCIC
																									X		SW-A
								X			X		X			X	X	X		X		X		X	X		RHR-A
								X			X		X			X	X	X		X		X		X	X		LPCI-A
							X																				SW-B
X	X	X	X	X			X		X																		RHR-B
X	X	X	X	X	X		D		X																		LPCI-B
																									X		SW-C
X			X		X	X				X		X										X		X	X		LPCI-C
X					X	X	X			X	X	X	X			X		X		X		X		X	X		LPCS
																											TSW-A
																											TSW-B
			X																							X	CAS-A
																										X	CAS-B
																										X	CAS-C

4.6.1 The Loss of Feedwater Sequence and Fire Event Trees

Any fire sequence which caused damage to the condensate, feedwater, or main steam systems was assumed to cause loss of feedwater, and was modeled using the loss of feedwater event tree, TF.

The event tree TF functional headings are arranged in the approximate timewise order that the functions occur. The functions represented by each heading for each fire area represented by the TF event tree is described in the following subsections:

FxTF - Initiating Event Frequency

The initiating event frequency is always set equal to 1.0. Therefore, the core damage frequency for each event tree is a conditional core damage frequency.

C-Scram

The water level decreases rapidly due to the mismatch between coolant inventory loss (steam) and supply (feedwater) until the low level SCRAM setpoint L3 is reached. The unavailability of the RPS/SCRAM function is assumed to be 1.40E-05/demand, taken from Reference 4.10.3. The failure to SCRAM combined with an initiating frequency of fire is very small and this branch of the event tree is not considered further.

M-SRVs Open

The reactor scram, resultant turbine trip, and continued loss of feedwater results in water level reaching L3 isolation which closes MSIVs if they were not already closed due to fire damage. This results in RPV pressure increase and lifting of the SRVs to maintain pressure control. The SRV failure to open probability is 5.00E-5/demand (Reference 4.10.3). The failure of SRVs to open combined with an initiating frequency of a fire is very small and this branch of the event tree is not considered further.

P-SRVs Reclose

The failure of an SRV to reclose is 1.91E-02/demand (Reference 4.10.3). The failure of an SRV to reclose is independent of the fire initiator and therefore, it is bounded by the SORV/IORV event evaluation in Reference 4.10.3.

U-HPCS or RCIC Available

When the water level reaches L3, in addition to the reactor scram, HPCS and RCIC initiation signals are generated. If HPCS and RCIC or their support systems are unaffected by the fire, the branch unavailability is calculated by fault tree linking of HPCS and RCIC. If HPCS and/or RCIC availability is affected by the fire, their affected system is assumed to be unavailable.

X-ADS Available

If the HPCS or RCIC is successful, this function and the V_1 (low pressure injection, see below) are not required and the success branch is drawn through these functional headings. If HPCS and RCIC fail to inject, the level drops until manual ADS is initiated. The ADS unavailability is calculated from its system fault tree. The EOPs prevent automatic ADS until water level reaches TAF, then Operator action is taken. Therefore, any direct or support system fire damage is taken into account.

V_1 - LPCS or LPCI Available

If ADS function is unsuccessful, the low pressure injection system, LPCS (1 train) and LPCI (3 trains) are not effective and the lower ADS branch line passes through this function directly to core damage end state. Upon successful ADS, the low pressure injection systems, LPCS and LPCI, are automatically or manually initiated for RPV water level control. The LPCS and LPCI unavailability are calculated from their system fault tree. Therefore, any direct or support system fire damage is taken into account.

V_4 - SW Crosstie To RHR-B Available

If the V_1-LPCS, LPCI function is successful an alternate low pressure injection system is not important and the branch line passes through this function. However, if LPCS, LPCI fail, then the alternate low pressure injection via SW-B crosstied to RHR-B line is an alternative path. If the fire damage renders SW-B unavailable, this function is assumed unavailable and nonrecoverable.

W_1 - One Loop of RHR Available

If alternate low pressure injection via SW-B crosstie is unsuccessful, the branch line passes through this decay heat removal function to a core damage end state. If however, any one of the injection paths (U, V_1, or V_4) are successful, then this functional heading is the first to determine if the decay heat can be successfully removed from the primary containment. This functional heading determines the availability of one of two RHR loops in suppression pool cooling mode and is calculated directly from the RHR system fault tree. Therefore, any direct or support system fire damage is taken into account. If this function is successful, the end state is no core damage and the branch line passes through the following functions of alternate means to remove decay heat.

Z - MSIVs Reopened and PCS Available

If the RHR is unavailable for decay heat removal, then the operators will attempt to reopen the MSIVs and establish the condenser as the heat sink. If the power conversion system is damaged by the fire, this functional unavailability is set to 1.0; otherwise, the unavailability is determined by the power conversion system fault tree which properly accounts for any support system fire damage.

W_2 - Containment Venting Available

If the reopening of MSIVs and establishing the power conversion system is successful, the branch line passes through this function to a non core damage end state. If that function is unsuccessful, the decay heat can be removed by controlled venting of the primary containment. If the containment venting is successful, the fire sequence ends in a no core damage end state. The unavailability of containment venting is determined from the venting system fault tree. Therefore, any direct or support system fire damage is taken into account. If venting and reopening of MSIVs/ PCS establishment are unsuccessful, core damage occurs with a likelihood of 0.33. This likelihood reflects the chance that injection fails following containment failure. This development is identical to that used in the IPE.

The loss of feedwater event tree used as the basis to quantify the conditional core damage frequency of the fire areas initiating a loss of feedwater is shown in Figure 4.6-1.

The quantification of the event tree for the fire areas is discussed in Section 4.6.3 and the results tabulated in Table 4.6-2.

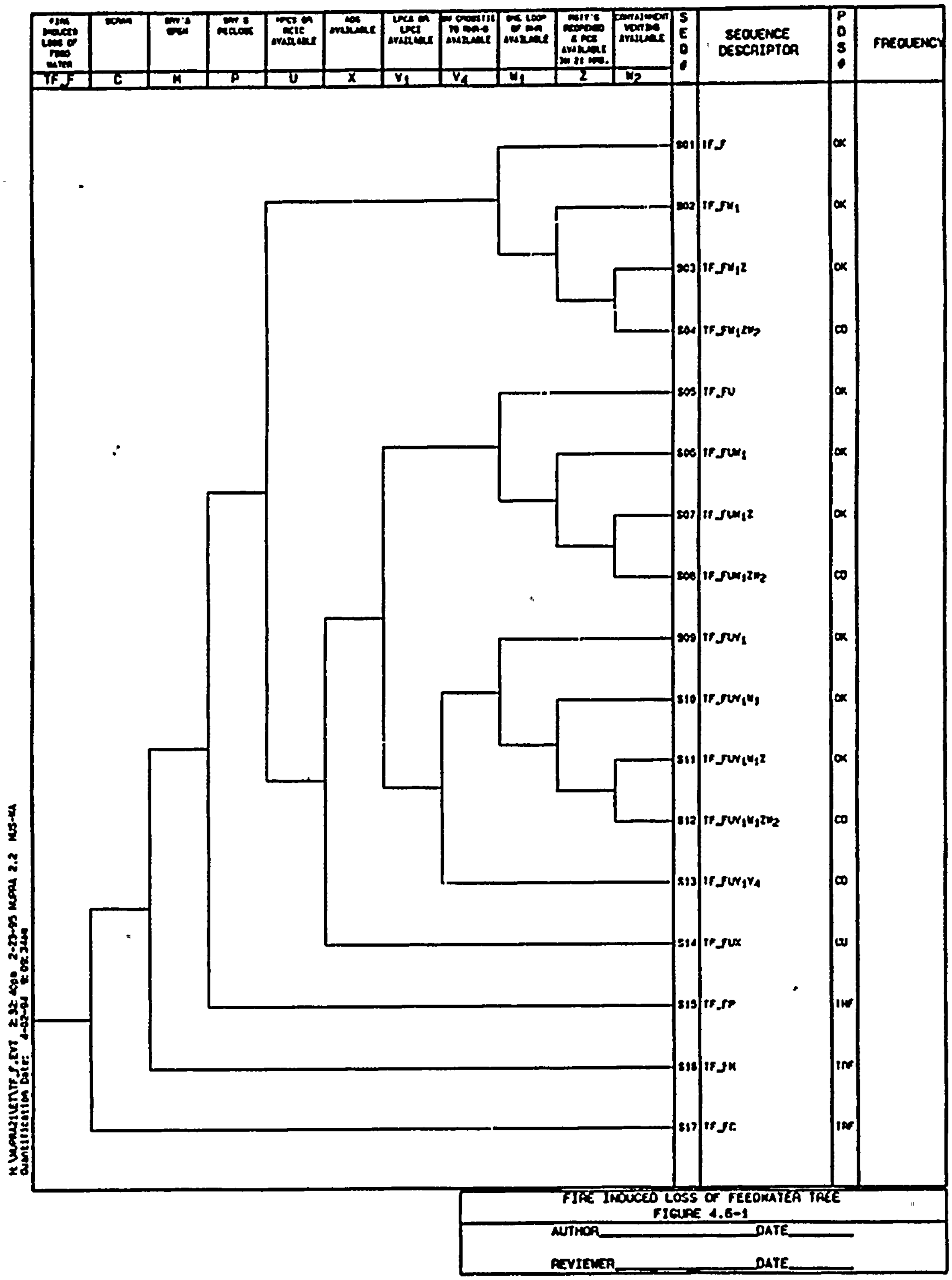


Figure 4.6-1
Loss of Feedwater Event Tree

4.6.2 The Turbine Trip Sequence and Fire Event Trees

The fire sequences that did not lead directly to loss of the power conversion system were modelled using the turbine trip event tree, TT.

The functional headings for the turbine trip event tree, TT, are the same as for the loss of feedwater event tree, TF, except the potential availability of the power conversion system and the potential availability of the condensate system for injection are included. (Comparison of Figure 4.6-1 to Figure 4.6-2 clarifies this difference.) These functional headings for TT are as described in Section 4.6.1 with the following additional functional headings.

Q - MSIV, COND, FRW and PCS Available

Following an initiating fire event that results in a reactor SCRAM and turbine trip, removal of steam to the condenser via turbine bypass valves with makeup to the vessel via feedwater/condensate is the normal post-SCRAM procedure. If this function is successful, the branch line passes directly to a no core damage end state. This functional unavailability is determined from the power conversion system fault tree. Therefore, any direct or support system fire damage is taken into account.

V_2 - Condensate or Booster Available

If the high pressure injection systems have failed their function, but ADS is successful, then low pressure injection is possible. The first low pressure injection function V_1 is the same as described in Section 4.6.1. If the LPCS/LPCI function fails, then the condensate and/or condensate booster pumps are potentially available. This functional unavailability is determined from the condensate system fault tree. Therefore, any direct or support system fire damage is taken into account.

The turbine trip event tree used as the basis to quantify the conditional core damage frequency of the fire areas not initiating loss of feedwater is shown in Figure 4.6-2.

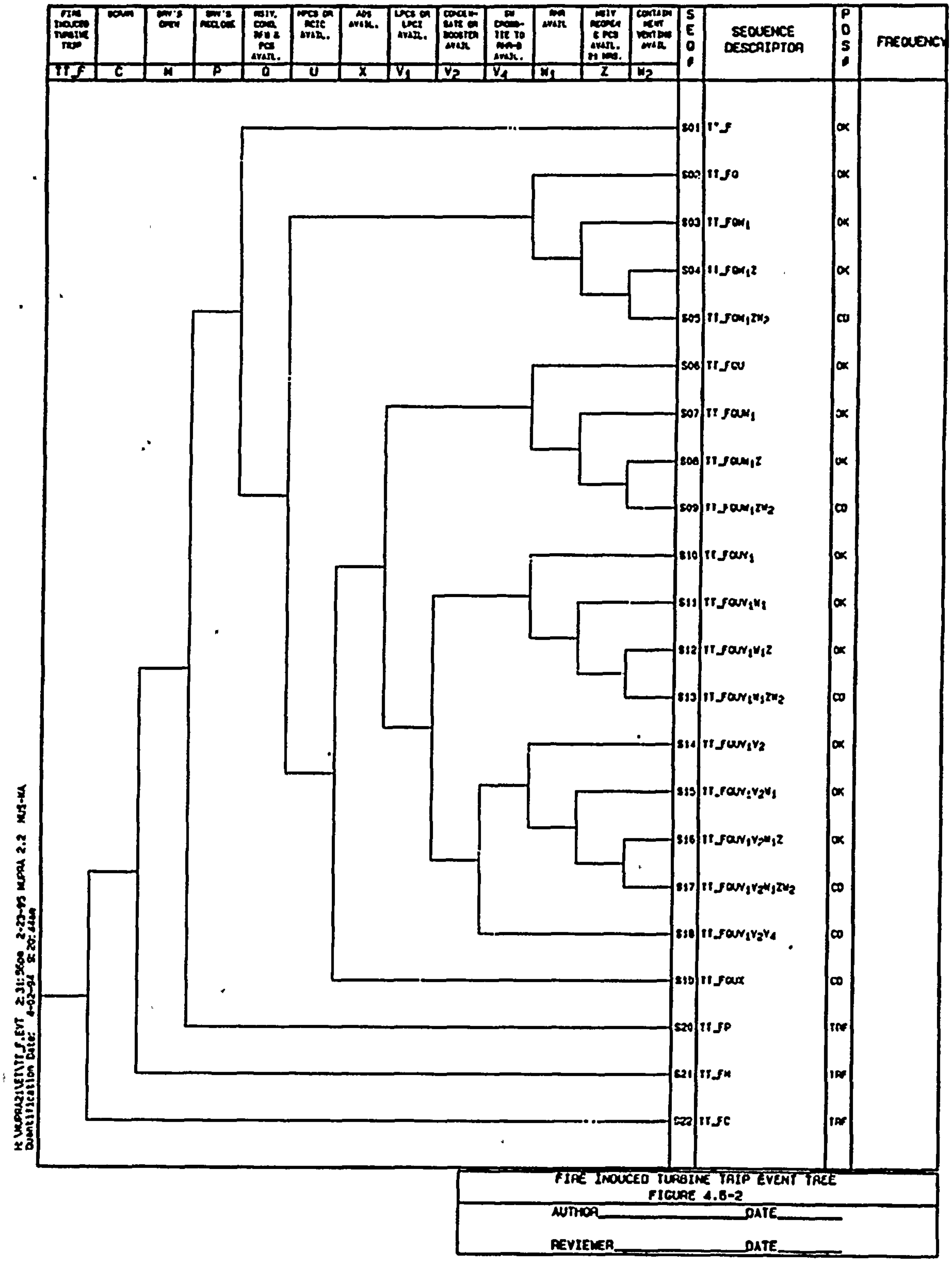


Figure 4.6-2
Turbine Trip Event Tree

4.6.3 Quantification of Sequence Frequencies

All systems were dependent, in some way, on other systems for successful operation. System dependencies were accounted for by explicitly including the dependencies within the models. This process is explained in more detail in revision 1 of the WNP-2 IPE submittal. Briefly, internal and external transfer gates within each fault tree model serve to connect each fault tree with its required supporting fault trees.

The event trees presented earlier in this section identify functional equations associated with each of the branch points. A functional equation represents the unavailability of one or more front-line systems and their support systems to perform a specific function within the conditions identified in the tree. The event tree conditions which impact the operation of systems are accounted for by the use of house events embedded in the fault trees. For the fire PSA quantification, two sets of house event update files were used. The first set is the base solution set described in revision 1 of the original IPE submittal for WNP-2. This set performs the function of accounting for event tree timing, like the presence or absence of RPV level signals, RPV pressure, etc.. The second set of house event files is fire area specific. This set performs the function of accounting for fire damage in a given scenario. Thus, at each branch point in the event tree, the supporting fault tree(s) are linked with its support systems, updated against the appropriate event tree timing house event file, then updated again against the fire area specific house event file before it is solved.

Event tree functional equations may be developed in one of three ways: 1) by solution of a single linked frontline system fault tree, 2) by a single basic event developed by the analyst (useful when a functional heading is known to be failed), and 3) by a boolean combination of two or more single linked front line system fault trees. The functional equations were solved with a truncation limit of 1E-8.

Following completion of the development of the functional equations which define the cutsets for each event tree heading, the accident sequence quantification is performed to develop the accident sequence frequencies. Each accident sequence is uniquely defined by the functional successes and failures along the accident sequence path through the event tree. The quantification of each accident sequence involves combining the functional equations for each functional failure with a boolean "AND" and performing a boolean reduction of the resultant equation. After the system equations are ANDed together, the failure equations for functional successes are deleted from this equation. Accounting for success paths is an essential step to avoid developing incorrect cutsets. One additional step is performed for each accident sequence, deleting the disallowed cutsets which arise due to modeling simplifications. An example of such a disallowed cutset is (HPCS maintenance * RCIC maintenance). These maintenance activities are not allowed to occur at the same time. A truncation level of 1E-9 was used for all accident sequence quantification steps.

After quantification of the fire event trees using initiator frequencies of 1.0 to obtain conditional core damage accident frequencies, the results were combined on a fire area by fire area basis with the ignition frequency and the detection and suppression failure rates to obtain the core damage contribution for each area. Table 4.6-2 shows the results of this initial combination for each fire area.

4.6-2 CORE DAMAGE FREQUENCY

PSA Fire Zone	Description	Representative Plant Initiator	Event Tree	Fire Initiating Event Frequency	Fire Suppression Unavailable	Conditional Prelim Core Damage Probability	Prelim Calculated Core Damage Frequency(1)
DG-1	HPCS DG Room	TT	F29	3.1E-02	0.025	5.91E-05	4.64E-08
DG-10	Deluge Valve Rm	NA	NA	NA	NA	NA	NA
DG-2	DG Room #1	TT	F28	3.1E-02	0.025	3.31E-04	2.60E-07
DG-3	DG Room #2	TF	F26	3.1E-02	0.025	2.25E-04	1.77E-07
DG-4	DG-1 Oil Pump Rm	TT	F28	2.3E-03	0.025	3.31E-04	1.88E-08
DG-5	DG-2 Oil Pump Rm	TT	F22	2.3E-03	0.025	1.19E-04	6.75E-09
DG-6	HPCS DG Oil Pump	NA	NA	NA	NA	NA	NA
DG-7	HPCS Day Tnk Rm	NA	NA	NA	NA	NA	NA
DG-8	DG-1 Day Tank Rm	TT	F20	7.0E-04	0.025	3.71E-05	6.45E-10
DG-9	DG-2 Day Tank Rm	TT	F25	7.0E-04	0.025	4.29E-05	7.45E-10
R-10	Elevator	NA	NA	NA	NA	NA	NA
R-11	Stairway	NA	NA	NA	NA	NA	NA
R-12	Elevator	NA	NA	NA	NA	NA	NA
R-15	Lobby	NA	NA	NA	NA	NA	NA
R-17	S Valve Room	TT	F19	1.6E-03	1	1.47E-05	2.35E-08
R-18	Div 2 MCC Rm	TT	F27	2.9E-03	1	5.20E-05	1.49E-07
R-19	H2 Recombiner Rm	TT	F19	2.9E-03	1	1.47E-05	4.22E-08
R-1A	Railroad Bay 441′	NA	NA	NA	NA	NA	NA
R-1B	NE Reactor Bldg 471′	TF	F7	1.6E-03	1	2.88E-02	4.61E-05
R-1C	SE Reactor Bldg 471′	TF	F4	1.1E-02	1	3.36E-05	3.60E-07
R-1D	NW Reactor Bldg 471′	TF	F7	2.9E-03	1	2.88E-02	6.31E-05
R-1E	SW Reactor Bldg 471′	TF	F2	3.1E-03	1	1.11E-05	3.49E-08
R-1F	NE Reactor Bldg 501′	TT	F24	2.9E-03	1	6.69E-05	1.92E-07
R-1G	SE Reactor Bldg 501′	TT	F23	2.9E-03	1	6.69E-05	1.92E-07
R-1H	NW Reactor Bldg 501′	TF	F6	2.3E-03	1	9.44E-04	2.13E-06
R-1I	SW Reactor Bldg 501′	TT	F18	7.9E-04	1	1.20E-04	9.42E-07
R-1J	Reactor Bldg 522′	TF	F6	8.2E-03	1	9.44E-04	7.69E-06
R-1K	Reactor Bldg 548′	TT	F23	1.4E-2	1	6.69E-05	9.37E-07

4.6-2 CORE DAMAGE FREQUENCY

PSA Fire Zone	Description	Representative Plant Initiator	Event Tree	Fire Initiating Event Frequency	Fire Suppression Unavailable	Conditional Prelim Core Damage Probability	Prelim Calculated Core Damage Frequency(1)
R-1L	Reactor Bldg 572′	TT	F23	1.4E-02	1	6.69E-05	4.65E-07
R-1M	Equipment Hatch	TF	F6	7.3E-03	1	9.44E-04	6.90E-06
R-1N	Stairwell (Uninstalled)	TT	F29	1.2E-03	1	5.91E-05	7.09E-08
R-1O	Stairwell S-3	TF	F7	1.6E-03	1	2.88E-02	4.75E-05
R-2	Drywell/Containment	NA	NA	NA	NA	NA	NA
R-21	S Valve Room	NA	NA	NA	NA	NA	NA
R-3	HPCS Room	TT	F29	5.0E-03	1	5.91E-05	2.93E-07
R-4	RHR-B Room	TT	F27	5.0E-03	1	5.20E-05	2.57E-07
R-5	RHR-A Room	TT	F18	5.0E-03	1	1.20E-04	6.05E-07
R-6	RCIC Room	TT	F25	2.8E-03	1	4.29E-05	1.21E-07
R-7	RHR-C Room	TT	F21	3.4E-03	1	3.10E-05	1.07E-07
R-8	LPCS Room	TT	F21	2.7E-03	1	3.10E-05	8.43E-08
R-9	Stairway	NA	NA	NA	NA	NA	NA
R-H22/P009	Instrument Rack Room	NA	NA	NA	NA	NA	NA
R-H22/P021	Instrument Rack Room	TT	F19	7.0E-04	1	1.47E-05	1.02E-08
R-H22/P027	Instrument Rack Room	TT	F27	7.0E-04	1	5.20E-05	3.61E-08
R-IR-73	Instrument Rack Room	NA	NA	NA	NA	NA	NA
RC-10	Main Control Rm	NA	NA		NA	NA	NA
RC-11	A A/C Room	NA	NA	NA	NA	NA	NA
RC-12	B A/C Room	NA	NA	NA	NA	NA	NA
RC-13	Emerg Chiller	NA	NA	NA	NA	NA	NA
RC-14	SWGR Rm #1	TF	F8	5.4E-03	1	6.67E-03	3.57E-05
RC-15	Stairway	NA	NA	NA	NA	NA	NA
RC-16	Stairway	NA	NA	NA	NA	NA	NA
RC-17	Elevator	NA	NA	NA	NA	NA	NA
RC-18	Stairway	NA	NA	NA	NA	NA	NA
RC-19	Corridor	TF	F31	7.0E-04	1	4.90E-03	3.41E-06
RC-1A	Radwaste Bldg 437′	TF	F11	2.9E-03	1	1.80E-04	5.17E-07

4.6-2 CORE DAMAGE FREQUENCY

PSA Fire Zone	Description	Representative Plant Initiator	Event Tree	Fire Initiating Event Frequency	Fire Suppression Unavailable	Conditional Prelim Core Damage Probability	Prelim Calculated Core Damage Frequency(1)
RC-1B	Radwaste Bldg 467′	TF	F3	1.2E-03	1	4.24E-07	4.96E-10
RC-1C	Radwaste Bldg 487′	TT	F17	1.1E-02	1	7.82E-07	8.84E-09
RC-1D	Equipment Hatch	NA	NA	NA	NA	NA	NA
RC-1E	Equipment Hatch	TF	F1	1.1E-03	1	1.01E-06	1.08E-09
RC-1F	Equipment Hatch	NA	NA	NA	NA	NA	NA
RC-20	Pipe Chase	TT	F18	6.96E-04	1	1.20E-04	8.35E-08
RC-2A	Cable Spread Rm	TF	F13	7.46E-04	0.025	3.69E-02	6.88E-07
RC-2B	Cable Spread Rm	TF	F10	7.46E-04	0.025	3.89E-02	7.25E-07
RC-2C	Combust Free Zone	NA	NA	NA	NA	NA	NA
RC-3	Cable Chase	TF	F16	7.1E-04	0.025	4.48E-02	7.96E-07
RC-4	Div 1 Elec Equip Room	TF	F32	9.2E-03	1	6.67E-03	6.14E-05
RC-5	Battery Room 1	TF	F30	2.5E-03	1	8.47E-04	2.20E-06
RC-6	Battery Room 2	NA	F34	2.5E-03	1	4.90E-03	1.23E-05
RC-7	Div 2 Elect Equip	TF	F33	9.2E-03	1	4.75E-03	4.35E-05
RC-8	SWGR Rm #2	TF	F14	5.4E-03	1	4.90E-03	2.63E-05
RC-9	Remote Shtdn Rm	TF	F5	1.0E-02	1	4.52E-03	4.61E-05
SW-1	SSW Pump House 1A	TT	F20	4.0E-03	1	3.71E-05	1.50E-07
SW-2	SSW Pump House 1B	TT	F25	4.2E-03	1	4.29E-05	1.79E-07
TG-1A	Turbine Gen West 44	TF	F9	4.3E-03	1	2.92E-02	1.27E-04
TG-1B	Turbine Gen Center 4	TF	F3	6.3E-03	1	4.24E-07	2.67E-09
TG-1C	Turbine Gen East 44	TF	F12	7.1E-03	1	1.21E-04	8.53E-07
TG-1D	Turbine Gen West 47	NA	NA	NA	NA	NA	NA
TG-1E	Turbine Gen East 47	NA	NA	NA	NA	NA	NA
TG-1F	Turbine Gen Center 4	TF	F3	7.0E-04	1	4.24E-07	2.95E-10
TG-1G	Turbine Gen West 50	NA	NA	NA	NA	NA	NA
TG-1H	Turbine Gen Center 5	NA	NA	NA	NA	NA	NA
TG-1I	Turbine Gen East 50	NA	NA	NA	NA	NA	NA
TG-1J	Equipment Hatch	TF	F3	4.1E-03	1	4.24E-07	1.76E-09

4.6-2 CORE DAMAGE FREQUENCY

PSA Fire Zone	Description	Representative Plant Initiator	Event Tree	Fire Initiating Event Frequency	Fire Suppression Unavailable	Conditional Prelim Core Damage Probability	Prelim Calculated Core Damage Frequency[1]
TG-1K	Turbine Gen Corr	TF	F15	7.11E-04	1	9.83E-02	6.99E-05
TG-2	Turbine Oil Storage	NA	NA	NA	NA	NA	NA
TG-3	Stairway	NA	NA	NA	NA	NA	NA
TG-4	Elevator	NA	NA	NA	NA	NA	NA
TG-5	Stair A3	NA	NA	NA	NA	NA	NA
TG-6	Stairway	NA	NA	NA	NA	NA	NA
TG-7	Hydrogen Seal Oil Rm	NA	NA	NA	NA	NA	NA
TG-8	Stairway	NA	NA	NA	NA	NA	NA
TG-9	Turbine Oil Reservoir	NA	NA	NA	NA	NA	NA

(1) Preliminary Calculated CDF = Init Event Frequ * Fire Suppression Unavail * Cond'al Prelim CDF

4.6.4 Important Fire Areas

Based on the preliminary CDF values presented in the last column of Table 4.6.2, the fire areas can be screened further so that recovery actions and/or analysis refinements are applied only to important fire areas. In general, a more realistic assessment will result in a factor reduction of 2 to 100 from the preliminary values. Therefore, only the fire areas with a preliminary CDF > 1E-6 are considered important. There are 16 fire areas that satisfy this criteria and they are discussed in the following subsections. The remaining fire areas contribute a cumulative CDF approximately 2E-5/yr. However, as discussed below for the important fire areas, this value is due to conservative assumptions and they are ignored from further discussion.

4.6.4.1 Fire Area RC-14: SWGR RM #1

A damaging fire in this area results in loss of safety-related AC bus SM-7 which results in the closure of MSIVs, loss of venting capability, and loss of RCIC, LPCS, RHR-A, and SW-A systems. Seventy (70) percent of the contributors to the initiation frequency is from the electrical cabinets, only three (3) percent contribution is from transient combustibles. Therefore, an automatic suppression system would be of minimal value to prevent loss of major components (cabinets) and limiting transient combustibles would, likewise, have minimal input.

Since the fire damage is to the support system (AC bus power) and not to the components of the power conversion system or to the vent system, there is a potential for recovery. On loss of SM-7 the operators have a predefined instruction/procedure to transfer RPS Bus A power to Alternate A power supply (Alt A). This allows the RPS to be reset and to reset the isolation signals. This allows reopening of the MSIVs, if closed, and recovery of the power conversion system. Likewise, the containment venting becomes available. Since the steps are proceduralized and the time available (approximately 21 hours) is sufficient, a recovery factor of 0.06 is applied (Reference 4.10.3). This factor is the most conservative between recovery of the power conversion system and recovery of the vent system derived in the IPE. This factor reduces the calculated CDF for the fire zone to 1.29E-07/yr.

4.6.4.2 Fire Area RC-4: Div 1 Electrical Equipment Room

A damaging fire in this fire area results in potential loss of Div 1 AC power panels (MC-7), Div 1 dc power (bus and battery charger), RHR-A, 250 Vdc power train 2/1, and isolation of MSIVs. Fire ignition sources are primarily electrical, low voltage sources (cabinets, MG sets, battery chargers). Transient combustibles contribute less than 2%. The Fire Hazards Analysis (Reference 4.10.4) allows transient combustibles in this room equivalent to 55 gallons of oil. There is no reason for fuel oil or lubricating oil to be in this room in that quantity. For smaller combustible quantities such as 30 pounds of class A combustibles, or 2 gallons of solvent, COMPBRN shows cable tray damage would be very unlikely. With this restriction, the probability for fire propagation to equipment other than the source, i.e., cabinet, MG set, or battery charger, is assumed to be 0.1. However, it was assumed that ambient room temperatures within the room in which the fire occurred would exceed the environmental limits for all cabinets located within the room. RC-4 is comprised of two rooms. One room, RPS room #1 contains the MG Sets and the resulting equipment failures assumed for this room is closure of the MSIVs. A recovery factor is applicable for re-opening the MSIVs as discussed in Section 4.6.4.1 above. The second room, Electrical Equipment room #1 contains battery chargers S1/1 and S2/1. Both of these chargers and the busses to which they are connected are assumed to be failed if the fire occurred in this room. Further, since the recommended restrictions on combustibles in this area makes it unlikely that significant damage would occur due to a transient combustible fire, it is reasonable to assume a reduction in Welding/Combustibles and transient combustibles of 0.1. Therefore, if transient combustibles are limited to 30 pounds of Type A combustibles or 2 gallons of solvent and the appropriate MSIV recovery factor applied, the calculated CDF is reduced to 5.54E-07/yr.

4.6.4.3 Fire Area RC-7: Div 2 Electrical Equipment Room

This is the Div 2 AC/DC equivalent of the fire zone discussed in Subsection 4.6.4.2 for Div 1. If the same restriction on transient combustibles is placed in this Fire Area and the appropriate MSIV recovery factor applied, the calculated CDF is reduced to 4.06E-07/yr.

4.6.4.4 Fire Area RC-8: SWGR Room #2

This is the Div 2 loss of SM-8 equivalent of the fire area discussed in Subsection 4.6.4.1 for Div 1. The recovery of SM-8 is proceduralized with sufficient recovery time. Using the same recovery probability results in a calculated CDF of 9.43E-08/yr.

4.6.4.5 Fire Area RC-9: Remote Shutdown Room

A damaging fire in this area results in the loss of Div 2 SM-8 power with consequential loss of RHR-B, SW-B, RCIC venting, and MSIV closure. The fire ignition frequency is almost all due to the electrical cabinets (93%) with only 1% due to transient combustibles.

Since the fire damage is to the electrical support system (SM-8 and RCIC power), the power conversion system is recoverable by reopening the MSIVs (see Subsection 4.6.4.1 for further discussion). Using the same recovery factor, the calculated CDF becomes 1.66E-7/yr.

4.6.4.6 Fire Area TG-1A: Turbine Generator Building West 441′

A damaging fire in this area causes loss of both 230 kV and 115kV offsite power sources to SM-7, loss of power conversion system, and consequential failure of RCIC. This fire zone was not modelled with COMPBRN, however, it is assumed to have similar combustible geometry characteristics as 471′ of the reactor building. The COMPBRN analysis shows that for the case of maximum combustibles (one 55 gallon drum of oil), the cables were not damaged regardless of whether ventilation is turned on or off. Furthermore, pump/valve fires also were demonstrated to not affect cables. Cabinet fires, on the other hand damaged overhead cables within a 1.2 meter radius.

Based on these findings, a 0.1 likelihood that pump, valve, ventilation system, transient combustible, and welding/combustible fires propagate and cause widespread cable damage within the area was included. Cabinet fires and welding/cable fires are conservatively assumed to cause widespread cable damage. With these refinements, the calculated CDF is 5.91E-07/yr.

4.6.4.7 Fire Area TG-1K: Turbine Generator Corridor

A damaging fire in this Fire Area results in loss of 230 kV and 115kV to SM-7, SM-8, and loss of DG-1, RHR-A, SW-A, HPCS, RCIC, SW-C, and LPI-C. Welding fires contribute 78%, transient combustibles 20%, and cables (self-igniting) contribute 2%.

This fire zone was modelled with COMPBRN. The analysis shows that for the case of maximum combustibles (one 55 gallon drum of oil), no cable damage will occur. Based on these findings, it was assumed that fires involving transient combustibles have a 0.1 likelihood of causing significant cable damage. Furthermore, the resident time of combustibles is a small fraction of the year. It is assumed that the time oil drums are present is 240 hours per year. The new initiating event frequency becomes: 9.41E-05/yr.

A portion of the TG Corridor cable trays is sprinkled and the 0.025 auto suppression factor can be credited for this portion. The calculated CDF becomes 2.91E-6/yr with these refinements.

4.6.4.8 Fire Area R-1O: Stairwell S-3

The fire damage from this area was assumed to be a 20 foot cylinder surrounding the full length of the stairwell. COMPBRN analysis shows the maximum combustibles (one 55 gallon drum of oil) would not result in damage on more than one floor. That is, a fire would not propagate up the stairwell to involve multiple floors. Since each floor area included in the cylinder is included in other Fire Areas, the contribution from this Fire Area is zero.

4.6.4.9 Fire Area RC-19: Div 1/Div 2 Corridor

A damaging fire in this area disables distribution panels for Div 1 DC power, CAS-B and closes the MSIVs. The contribution to the initiating event frequency was 80% from welding initiation and 20% from transient combustibles.

In developing a fire scenario, an ignition source must be postulated due to some cause (e.g., welding) and propagate to the postulated 55 gal drum of oil. For the corridor region in this fire area, there is no permanent storage of oil drums, and the transient time (i.e., time oil drums may be present) is assumed to be 240 hours per year. If the propagation of welding caused ignition is assigned a probability of 0.1, then the calculated CDF for this area is 1.06E-6/yr.

4.6.4.10 Fire Areas RC-5 and RC-6: Div 1 and Div 2 Battery Rooms

A damaging fire in RC-5 or RC-6 disables dc power divisions 1 and 2, respectively. Decay heat removal via the power conversion system was assumed to be unavailable. However, the WNP-2 IPE (Reference 4.10.3) shows that there is an 88% probability that PCS will survive if one division of dc power is disabled. Therefore, applying a factor of 0.12 for PCS unavailability, the refined CDFs for RC-5 and RC-6 are 2.55E-07/yr and 1.48E-6/yr, respectively.

4.6.4.11 Fire Area RC-9: Remote Shutdown Room

A damaging fire in this zone disables much of division 2 AC support power. Refinement of the CDF for this area was treated in a manner similar to that for RC-14 as discussed in Section 4.6.4.1. The refined CDF is 1.66-E-7/yr.

4.6.4.12 Fire Areas R-1B, R-1D, R-1H, R-1J, and R-1M: Reactor Building

Damaging fires in these areas were found to result in calculated CDFs above 1E-6/yr.

COMPBRN analyses show that for the case of maximum transient combustibles (one 55 gallon drum of oil), the cables were not damaged regardless of whether ventilation is turned on or off. Furthermore, pump/valve fires also were demonstrated to not affect cables. Cabinet fires, on the other hand damaged overhead cables within a 1.2 meter radius.

Based on these findings, a 0.1 likelihood that pump, motor, valve, transient combustible, and welding/combustible fires propagate and cause widespread cable damage within the area. Welding/ cable fires are conservatively assumed to cause widespread cable damage.

With these refinements, the calculated CDFs are as follows:

	CDF
R-1B	7.35E-08/yr
R-1D	4.87E-07/yr
R-1H	2.90E-07/yr
R-1J	2.91E-07/yr
R-1M	3.77E-07/yr

4.6.5 Dominant Accident Sequences

The sixteen important fire areas and their calculated CDFs, discussed in the previous section are summarized in the following table:

Summary of Important Fire Areas With Refined CDF

Fire Area	Original CDF (1/yr)	Refined CDF (1/yr)	% Contribution for Refined CDFs
RC-14(F08):SWGR Rm #1	3.57E-05	1.29E-07	1
RC-19(F31):Div 1/Div 2 Elec/Batt Rm Corridor	3.41E-06	1.06E-06	12
RC-4(F32):Div 1 Elect Equipment Room	6.14E-05	5.54E-07	6
RC-5(F30):Div 1 Battery Room	2.20E-06	2.55E-07	3
RC-6(F34):Div 2 Battery Room	1.23E-05	1.48E-06	16
RC-7(F33):Div 2 Elect Equipment Room	4.51E-05	4.06E-07	4
RC-8(F14):SWGR Rm #2	2.63E-05	9.43E-08	2
RC-9(F05):Remote Shutdown Room	4.61E-05	1.66E-07	2
TG-1A(F09):Turb Gen Bldg West 441'	1.27E-04	5.91E-07	6
TG-1K(F15):Turb Gen Corridor	6.99E-05	2.91E-06	32
R-1B(F07):NE Reactor Building 471'	4.61E-05	7.35E-08	1
R-1D(F07):NW Reactor Building 471'	6.31E-05	4.87E-07	5
R-1H(F06):NW Reactor Building 501'	2.13E-06	2.90E-07	3
R-1J(F06):Reactor Building 522'	7.69E-06	2.91E-07	3
R-1M(F06):Equipment Hatch	6.90E-06	3.77E-07	4
R-1O(F07):Stairwell S-3	4.75E-05	0.0	0
TOTAL	6.01E-04	9.16E-06	100

The dominant sequences for several of the important fire areas are listed in Table 4.6-3. All of the dominant sequences for fires in the Radwaste Building show a common theme. Namely, the fire rendors venting (W_2) and power conversion system (Z) unavailable and damages one train of RHR or SW, such that the other decay heat removal train (W_1) unavailability dominates the sequences. It was noted that the cutsets for the unavailability of one RHR loop contained significant contribution from basic events related to the system being unavailable due to testing or maintenance.

Given the significant time available for initiation of decay heat removal, minimum of 24 hours, restoration or repair could be credited for these (Class 2) failure sequences. These recovery actions would only be applied to random failures or unavailabilities due to maintenance and testing and would not apply to equipment failed by fire.

The probability of recovering or repairing equipment could be determined using the time available to recover the equipment prior to core damage and an assumed log mean time required to perform the repair. The time available to recover the component was derived from Modular Accident Analysis Program (MAAP) analyses performed in support of the internal events PSA (Reference 4.10.3). A non-recovery probability could then be generated based on a simple exponential model for repair and a mean time to recover mechanical and electrical equipment from WASH-1400 (Reference 4.10.7). This model is expressed as; repair failure probability $P_f = e^{-t/T}$, where t = time available for repair and T = mean time to repair.

The mean time to repair varies with type of component. Electrical repairs, on average, require less time to perform than mechanical repairs. This correlates to a lower probability of failing to complete the average electrical repair than failing to complete the average mechanical repair during any given time period. Most sequences contain a large number of cutsets with mechanical failures or contain cutsets with mechanical failures that contribute significantly to the sequence total. To simplify the application of recovery/repair factors, all non-recovery probabilities could be conservatively based on the time required to repair mechanical failures of pumps. Only one recovery/repair factor would then be applied to each cutset.

The recovery factors described in Section 4.6 for the dominant fire areas provided sufficient core damage frequency reduction and the equipment repair factor was not used in the current analysis.

The fire zones in the Turbine Generator Building and Turbine Generator Corridor exhibit the same loss of decay heat removal sequence (W_1 Z W_2) as discussed for the Radwaste Building fire zones. In addition, each of these fire areas exhibit sequences in which high pressure injection (U) and low pressure injection (V_1 & V_4) are lost which makes loss of decay heat removal immaterial.

Table 4.6-3 Fire Area Dominant Sequences

FIRE AREA DOMINANT SEQUENCE	UNCORRECTED SEQUENCE CONDITIONAL FREQUENCY	SEQUENCE NAME
RC-4/F08TF-S04	5.80E-03	Initiator W_1 Z W_2
RC-7/F14TF-S04	4.40E-03	Initiator W_1 Z W_2
RC-6/F34TF-S04	4.48E-03	Initiator W_1 Z W_2
RC-19/F31TF-S04	4.48E-03	Initiator W_1 Z W_2
TG-1A/F09TF-S04	2.47E-02	Initiator W_1 Z W_2
TG-1K/F15TF-S14 F15TF-S13 F15TF-S08 F15TF-S12	7.40E-02 2.01E-02 3.29E-03 9.08E-04	Initiator U X Initiator U V_1 V_4 Initiator U W_1 Z W_2 Initiator U V_1 W_1 Z W_2

4.6.6 Control Room Fire Evaluation

The WNP-2 Control Room is analyzed as a separate fire area from the other fire areas in the Radwaste Control Building. This is justified since fire spread to the control room is highly unlikely due to:

1. Cables into and in the control room are qualified (fire resistant).
2. The cabling beneath the control room floor are protected by Halon fire suppression system.
3. The control room is maintained at a positive pressure which precludes smoke infiltration and retards fire propagation.
4. The control room is always occupied.

4.6.6.1 CR Fire Methodology

The general methodology used for evaluation of control room fires follows the approach discussed in Section 4.0 for the other fire areas. However, it differs in two respects:

1. Detailed fire propagation studies or analysis was not performed. It was assumed that cabinet fires in the control room do not spread beyond the confines of the cabinet provided the cabinet has solid metal or fire resistant boundaries. This assumption is supported by a) the Sandia cabinet fire tests in which all test fires in cabinets self-extinguished, and b) by the EPRI Fire Data Base (Reference 4.10.10) reports involving control room fires.

2. Regarding control room evacuation, two scenarios were developed for each cabinet fire, namely:

   Scenario 1 - The cabinet fire destroys all electrical components within the cabinet but does not propagate beyond this point. The control room is not evacuated unless a significant amount of control is lost that could be restored by manning the remote shutdown panel.

   Scenario 2 - The cabinet fire propagates beyond the electrical cabinet to such an extent that the control room must be evacuated and the remote shutdown panel must be manned to provide safe shutdown.

Control room fire scenarios were postulated by analyzing the effects from fires occurring in the electrical cabinets located within the room. A list of control room cabinets showing the equipment associated with each was reviewed to determine which PSA systems modelled in the Level 1 WNP-2 IPE (Reference 4.10.5) would be affected by a fire occurring in each cabinet. An initiating event frequency was obtained by dividing the initiating event frequency for control room cabinet fires (Reference 4.10.2) by the number of cabinets located in the room, 9.5E-3/yr/82 cabinets = 1.16E-4/yr. For the control room evaluation, it is assumed that all of the equipment in the cabinet where the fire originates will fail. This is bounding based on the data presented in the fire events data base. The control room cabinets are separated by solid metal barriers, so no fire propagation is assumed.

Generally speaking, the conditional probability for the occurrence of Scenario 1 was taken to be 0.97 (Reference 4.10.9). There was one exception to the use of the 0.97 probability. A probability of assured evacuation, or 1.0, was used if it was found that use of the remote shutdown panel could afford a greater degree of control of safety systems in comparison to the control available in the control room. This judgement was made by comparing the conditional core damage probability (CCDP) that would be expected if operators remained in the control room with that in which the operators evacuate. The conditional probability of Scenario 2 was taken to be the split fraction of the conditional probability of Scenario 1. Specifically, in most instances Scenario 2 occurs with a (1 - 0.97) = 3E-2 probability. If the probability for Scenario 1 was 0.0, of course that for Scenario 2 was 1.0.

4.6.6.2 CR Fire Assumptions

The general assumptions used for the control room fire analysis are as follows:

A. The impact from fires involving transient combustibles were assumed to be insignificant in relation to cabinet fires and were therefore not quantified. Such fires were judged to be typically of smaller magnitude and to be more readily extinguishable.

B. The control room was assumed to be evacuated for cabinet fires that result in loss of equipment control that exceeds that which is normally available in the remote shutdown panel. In other words, if it was found that less control would be available by remaining in the control room, it was assumed that the operators would evacuate and operate from the remote shutdown panel.

C. Each postulated control room cabinet fire was assumed to terminate control capability of all components associated with the cabinet and these components were assumed to fail in a state having the maximum impact on system or train unavailability. Control of such components was assumed to be restored from the remote shutdown panel if control is normally available from the shutdown panel.

4.6.6.3 CR Fire Analysis

The control room fire analysis was performed using the same approach as for the other fire areas. That is, the components assumed failed for each cabinet fire were determined based on the cable and instrument data for that cabinet from the WNP-2 cable data base (Reference 4.10.6). The equipment failed then was used to quantify the appropriate event trees for a conditional core damage probability (CCDP). The manner of quantifying the event trees used to produce the CCDPs is the same as that used for the other fire areas.

The core damage frequencies for the occurrence of fire for each cabinet was then obtained by multiplying initiating frequency, probability for control room operation/evacuation, and CCDP. The core damage frequency contribution for control room fires overall was obtained by summing each of the core damage frequency contribution for each cabinet.

4.6.6.4 CR Fire Results

The core damage frequency contribution calculated for control room fires is 8.4E-6/yr. The contribution to overall core damage frequency for individual cabinets is relatively uniform, with no particular outliers evident. The result of 8.4E-6/yr itself indicates that no particular vulnerabilities are evident in that it contribution is in line with those of other areas of the plant.

4.7 Analysis of Containment Performance

Two facets of containment performance were evaluated with regard to fire-induced damage. The impact of fires on containment structural performance and containment isolation or bypass was investigated.

The containment at WNP-2 is a free-standing steel pressure vessel design and is normally inerted with nitrogen. Because the containment contains minimal combustible material and is inerted during power operation, a significant fire within the containment is not expected to occur. The spaces surrounding the containment also contain very little combustible material and fire spread between these compartments is not credible. Therefore, because any fire in the spaces adjoining the containment will be contained within a single compartment and will be of limited duration and intensity, structural damage to the containment is not expected.

The potential for containment isolation or bypass was also investigated. Double isolation valves are provided on lines penetrating the containment that open to the free space of the containment. Closure of one of the valves in each line is sufficient to maintain the integrity of the containment boundary.

Fires can affect containment isolation valves in several ways: (1) failure of power cables or failure of motive power to solenoid-operated valves or air to air-operated valves will cause the valve to fail closed; (2) hot shorts in control cables to air-operated or solenoid-operated valves could possibly cause inadvertent valve opening; (3) failure of power cables to a motor-operated valve will fail the valve in its current position; and (4) hot shorts of control or power cables to a motor-operated valve could potentially result in a change of the valve's position. All of these, however, are probabilistically insignificant for the following reasons:

1. Many of the valves that connect the containment atmosphere to the reactor building are air-operated valves. With the exception of the wetwell-to-reactor building vacuum breakers, all the valves fail closed on a loss of air or power. Although extremely unlikely, if a hot short in one of these valve circuits were to occur that did not fail the protective fuse, manual recovery by removing fuses in the affected circuit would cause the valve to fail closed. The wetwell-to-reactor building vacuum breakers require differential pressure to open, and then only to allow flow into the wetwell.

2. Similar to the control circuits on air-operated and solenoid-operated valves, it is unlikely that a hot short in motor-operated valve circuits could occur without actuating the circuit's protective features, such as fuses. In addition, normally open motor-operated valves typically are in series with a closed isolation valve. The integrity of these piping systems would be unaffected by the fire. Low pressure systems with motor-operated valves that connect to reactor coolant piping either include at least one check valve in series with the motor-operated isolation valves or have two normally closed motor-operated valves. Therefore, containment bypass due to fire-induced spurious operation of motor-operated valves would require concurrent piping failure, simultaneous operation of the motor-operated valve, and/or failure of other valves.

For the reasons discussed above, fire-induced degradation of containment performance is expected to be negligible. There were no unique containment failure modes identified during the fire IPEEE analysis that differ from those identified in the internal events PSA.

4.8 Sandia Fire Risk Scoping Study Evaluation

Sandia National laboratory, as part of their Fire Protection Research Project, undertook two tasks in what is now referred to as the Fire Risk Scoping Study, NUREG-CR/5088 (Reference 4.10.8):

1. Review and update the perspective of fire risk in light of the information developed through the Fire Protection Research Project.

2. Identify and perform initial investigations of any potential unaddressed issues of fire risk.

Sandia reviewed four previously completed fire Probabilistic Risk Assessment (PSAs). The PSA risk scenarios were requantified using the data and information from the Fire Protection Research Project as a basis and included plant modifications made in response to implementations of Appendix R requirements at the plants under study. In performing the second task, Sandia developed a list of issues which they felt represented potential contributors to fire risks that had not been adequately addressed in previous risk assessments. Sandia concluded from these reassessments that fire may represent a dominant contributor to plant core damage risk and that these five issues should be addressed in future risk assessments.

The draft Sandia report was made available to several plant designers, fire researchers, industry representatives, fire protection consultants and regulators. They were asked to review the report and to ensure that, as far as practical, the list of unaddressed issues was complete. The most important industry response, provided to Sandia by the Edison Electric Institute Fire Protection Committee, was that "these issues are unaddressed by the selected method of risk evaluation and do not (necessarily) represent unaddressed risk issues for nuclear plants... this document is a report on the inadequacy of current risk assessment and research methodology for fire. There is no basis presented to indicate that regulatory requirements or implemented levels of fire protection are inadequate." Nonetheless, the list of five issues remains.

The NRC staff has requested the following five issues be addressed, in any future fire evaluation methodology:

1. Seismic/Fire Interactions
2. Fire Barrier Qualifications
3. Manual Fire Fighting Effectiveness
4. Control Systems Interactions
5. Improved Analytical Codes

Summaries of responses regarding each of these issues are addressed below.

4.8.1 Seismic/Fire Interactions

The seismic/fire interaction issue was addressed by a team consisting of a Supply System Analyst, an Earthquake Expert from EQE International and a Fire Protection Expert from VECTRA. The objective of the review was to provide a qualitative identification of design details associated with the fire protection systems which might be vulnerable to a seismic event. Several areas of the plant were reviewed and inspected for seismic/fire interaction issues involving both the fire protection systems and plant design features. Specifically, issues associated with seismic induced fire initiation, seismic induced actuation of fire suppression systems, and seismic induced degradation or diversion of suppression systems were reviewed.

The evaluations were conducted by plant walkdowns, review of design documentation and Appendix R documentation, and by interviews of fire protection staff. The study was focused on area which contained concentrations of safety related equipment, concentrations of combustibles, and locations of important fire protection components.

4.8.1.1 Seismic Induced Fire Initiation

Visual reviews of potential fire ignition sources and combustible inventories were performed for seismic vulnerability. In general, nonnuclear safety related plant components in safety related areas are designed for seismic loadings. Specific examples include non-class 1E electrical equipment such as MCCs, switchgear, and distribution panels which were reviewed and have anchorage comparable to safety related components.

Samples of combustible and transient inventories were reviewed during the area walkbys. Areas reviewed included the Turbine General Seal Oil Room which contained gaseous hydrogen piping, several areas with large cable concentrations and storage areas for transient combustibles. No unusual or unique seismic vulnerabilities were observed.

4.8.1.2 Seismic Induced Actuation of Fire Suppression Systems

Several types of fire suppression systems are used at WNP-2. These include both dry preaction sprinklers and wet sprinklers, Halon and CO_2 discharge systems.

4.8.1.2.1 Water Suppression Systems

All water suppression systems in areas of the plant containing safety related equipment were designed for seismic loadings. Piping and hanger details which were reviewed consistently incorporated structural detailing which was substantially more rugged than those utilized in commercial fire protection systems designed to National Fire Protection Association (NFPA) 13 seismic requirements. For example, piping and deluge valve stations appear to be well supported. Seismic Category I design criteria was applied to fire protection systems installed in Seismic Category I areas of WNP-2.

The non-safety Seal Oil Room in the Turbine Generator Building is protected by a deluge water system and was reviewed. No unusual or unique seismic vulnerabilities were observed.

All preaction systems reviewed were configured with Pyrotronics System 3 fire detection and control equipment. A review of these systems indicated they are of good industrial quality. The system controls are backed up with critical DC power supplies. The only vulnerabilities noted were that control equipment enclosures, conduit and junction boxes in the control systems are not sealed against water intrusion. Instances where fire detection and control equipment is located in areas with non-safety water systems were noted. Seismic induced failures of non-safety water systems could provide a mechanism for water intrusion into fire detection and control equipment and could result in system actuation.

Most fire suppression systems require redundant sensor/relay circuit closure to result in actuation of deluge valves and local head actuation for water release. In the event of an inadvertent actuation due to water intrusion into the fire control equipment, local sprinkler head actuation due to shaking and/or impact with other plant features would also be required to result in water discharge. These conditions do not appear to be widespread and would require several redundant levels of protection to be degraded before equipment in the local areas would be adversely affected.

4.8.1.2.2 CO_2 and Halon Systems

Halon suppression systems are provided locally for protection of control room electrical cabinets and cable routings. Control systems and Halon reservoirs are provided for each cabinet or area protected. Based on interviews with Supply System personnel, the system controls are backed up with critical DC power supplies. Control circuitry are of modem design, are reasonably robust and are mounted on Seismic Category I control room cabinetry. Analysis of relay components susceptible to chatter during the Review Level Earthquake (RLE) revealed that the Fire Protection system contains nine relays of unknown manufacturer and model number (i.e., unknown ruggedness). Should these relays chatter during RLE, the HALON System in the Control Room may spuriously actuate. This event could erroneously lead Control Room personnel to interpret that a fire is present. This spurious HALON injection scenario was confirmed by a circuit analysis.

The increase in plant risk attributable to the potential chattering of these FP relays was not considered significant enough to warrant further design or qualification efforts. However, since this scenario could occur during an already stressful sequence of events (earthquake), and since the actual seismic capability of these relays is in question, it was considered prudent to recommend that the Control Room operators be advised to take further steps to confirm an actual fire exists before accepting the HALON actuation at face value.

CO_2 suppression systems are provided for protection of the turbine generator. The CO_2 tank by Cardox was reviewed and appears to have some anchorage. The tank skid has visible anchor bolts which would be subject to bending. Attachment of the tank to the skid could not be determined. The Seal Oil Room in the Turbine Generator Building is protected by the CO_2 system and was reviewed. No unusual or unique seismic vulnerabilities were observed.

4.8.1.3 Seismic Induced Degradation or Diversion of Fire Suppression Systems

4.8.1.3.1 CO_2 Halon Systems

During the walkdown, there were no unusual or unique seismic vulnerabilities identified associated with the Halon system protecting the safety related control and electrical distribution systems. The CO_2 system for the non-safety Turbine Generator Building are reasonably well supported and seismic failures would not be expected until the SSE was significantly exceeded.

4.8.1.3.2 Water Systems

The Fire Protection Water System reviewed included the two diesel powered and two electric powered fire pumps, associated control cabinets and equipment, the above ground reservoir and a sample of the safety and non-safety related fire zones protected by the system.

The electric and diesel powered fire pumps appear to be well designed and anchored. These types of equipment are typically robust and perform well during earthquakes. The pumps are housed in two well constructed steel framed structures with slab on grade foundations for the pumps. Control panels for the diesel fire pumps are well anchored. Specific vulnerabilities for the diesel fire pumps noted were that the starting batteries were not restrained to their mounting stands. This could result in movement and potential overturning of the batteries with seismic ground motions high enough to overcome static friction. Overturning of batteries could result in failure of the diesel pumps to start.

Other vulnerabilities of the fire water system noted are that the system is designed with common pumps and delivery headers to protect both the safety related areas inside the main power block and non safety areas outside the power complex. The fire water system services some 15 to 20 general purpose buildings and plant facilities. These include structures with limited seismic capacities such as two story modular buildings and large yard transformers. The ability of the fire water system to deliver adequate flow to any, post seismic, fire demands inside the plant safety related areas could be compromised due to failures of fire water lines into some of the non safety structures. Actions by the plant fire brigade would be required to isolate any damaged portions of the systems.

4.8.1.4 Conclusions

Based upon our review of the fire protection systems, there are no sources of seismic induced fire initiation at reasonable levels of earthquake beyond the design basis. For inadvertent actuation, it was noted that in areas where safety related equipment is located, the fire protection systems are preaction and require redundant sensor/relay circuit closure to result in actuation of deluge valves and local head actuation for water releases. In the Control Room, it was determined that an increased risk of HALON actuation exists due to the unknown ruggedness of relays in the system. However, the increase in plant risk due to this event was not deemed to be great enough to warrant further action. Some vulnerabilities were noted regarding the availability of the fire protection systems, namely the support of batteries for the diesel driven fire pump and the fact that common pumps and headers are used to service both safety and non-safety areas. This is only a safety issue if a fire is initiated as a result of the earthquake, which as noted above, is not a probable event.

4.8.2 Fire Barrier Qualifications

Fire barriers and fire resistive construction prevent the spread of fire from one plant location to another. Design features of fire barriers prevent a fire from involving multiple fire areas before they are detected and extinguished. In order for fire barriers to be effective in the prevention of fire propagation, they must be maintained to their respective fire ratings. This is ensured by periodic inspection and maintenance of barriers. This periodic inspection and maintenance ensures that fire barriers will provide protection against fire propagation. WNP-2 is divided into multiple fire areas, separated from each other by fire barriers. These fire barriers include fire doors, fire dampers, penetration seals, and other rated assemblies. Section F.2.2 of Appendix F of the FSAR (Reference 4.10.4) describes the WNP-2 plant fire areas and barriers.

4.8.2.1 Fire Barrier Maintenance

Fire rated assemblies are inspected and verified operable per WNP-2 plant procedure. The purpose of this procedure is to perform an 18 month inspection and operational verification of:

at least 10% of the accessible Essential Fire Rated Assemblies listed in the Penetration Seal Tracking System (PSTS) Program

100% of all accessible Essential Fire Rated walls, ceilings, and floors

100% of all accessible Essential Fire Rated Raceway Enclosures

Penetration seals are installed, maintained and repaired in accordance with WNP-2 plant procedure. The purpose of this procedure is to provide detailed WNP-2 specific instructions for installation, maintenance and repair of penetration seals. The Penetration Seal Tracking System (PSTS) Program and drawing S1150 lists all penetration seal data for surveillance purposes.

Fire rated assemblies associated with the PGCC modules in the main control room are inspected in accordance with plant procedure. The purpose of this procedure is to provide guidance for performing an inspection of Fire Rated Assemblies in the main control room once per 18 months.

Fire dampers are inspected annually per WNP-2 plant procedure. The purpose of this procedure is to provide guidance for performing operability testing of Fire Dampers as required by NFPA 90A.

Fire doors are inspected weekly and annually per WNP-2 PPM 15.1.2, "Fire Door Operability." The purpose of this procedure is to provide guidance for performing operability testing of Fire Doors as required by NFPA 80.

In addition, fire rated raceway enclosures are installed, maintained and repaired in accordance with WNP-2 plant procedure. The purpose of this procedure is to provide detailed WNP-2 specific instructions for installation, maintenance and repair of fire rated raceway enclosures.

These procedures provide the prerequisites, precautions and limitations, materials and equipment, procedure, acceptance criteria, necessary forms, and a discussion to ensure that the inspections, tests and repairs are performed in the correct manner. All documentation is verified and maintained in accordance with the Volume 1 Series of Plant Administrative procedures.

4.8.2.2 Compliance with Regulatory Guidance

WNP-2 utilizes 3M Interam and Thermal Science, Inc. (TSI) Thermo-Lag 330-1 fire-rated raceway enclosures where fire areas contain redundant safe shutdown divisions. NRC Bulletin 92-01, Supplement 1, "Failure of Thermo-lag 330 Fire Barrier System to Perform Its Specified Fire Endurance Function" mandated that Licensees declare all Thermo-Lag 330-1 inoperable based on results of recent fire testing. Therefore, all Thermo-Lag raceway enclosures are considered inoperable and are under fire tour. A project was initiated at WNP-2 to restore Thermo-Lag operability or otherwise satisfy safe-shutdown regulatory requirements. Fire tours will be maintained until raceway enclosure operability is restored.

WNP-2 declared all fire-rated penetration seals inoperable in early 1994. This was mainly due to lack of documentation to show that all WNP-2 seal designs are qualified by adequate fire testing, similar to that discussed in NRC Information Notice 88-04, "Inadequate Qualification and Documentation of Fire Barrier Penetration Seals". Therefore, all fire-rated penetration seals are also under fire tour. A separate project was initiated at WNP-2 to restore penetration seal operability. Fire tours will be maintained until penetration seal operability is restored.

The surveillance procedure for raceway enclosures and penetration seals was cancelled. After penetration seal and/or raceway enclosure operability is restored, an improved revision of the surveillance procedure will be reissued and periodic fire barrier surveillances will resume.

WNP-2 has addressed NRC Information Notice 89-52, "Potential Fire Damper Operational Problems." This was addressed internally and noted that WNP-2 uses curtain type fire dampers to isolate fire protection zones. These dampers are tested via an open access hatch under air flow. WNP-2 secures all area fans upon call out of Fire Brigade (confirmation of fire). The Fire Brigade is trained in accordance with Prefire Plans to open selective dampers and restart fans for smoke removal.

WNP-2 has addressed NRC Information Notice 83-69, "Improperly Installed Fire Dampers At Nuclear Power Plants." This was addressed internally and noted that "Inspections of fire dampers were underway at WNP-2 before this Notice was issued late in October" [Reference letter dated September 29, 1983, G.C. Sorensen to NRR (G02-83-875)]. Information from this Notice has been included in the inspections that are still underway, and fire dampers are being upgraded as needed.

4.8.3 Manual Fire Fighting Effectiveness

The Supply System fire brigade is staffed, supplied, and trained per procedures and process developed by the Supply System in accordance with NRC guidance and National Fire Protection Standards. Fire fighting equipment required to support the fire brigade is also stocked and maintained per plant procedures developed by the Supply System in accordance with applicable standards.

The fire brigade at WNP-2 is composed of at least five trained people on each shift. Each brigade member is trained on all aspects of fire fighting as well as brigade assembly points and the responsibilities of each member. The brigade leader and at least two other members on each brigade shift are knowledgeable in plant systems and operations. The balance of the fire brigade is composed of trained support personnel including at least one health physics technician. The minimum operating shift crew complement required to safely shutdown the unit is in addition to the personnel composing the fire brigade on any shift.

In addition to initial classroom training and quarterly update training, practice sessions and drills are conducted regularly. Fire brigade members receive hands-on structural fire fighting training at least once a year to provide experience in actual fire extinguishment and the use of emergency breathing apparatus and personal protection equipment. Drills are conducted at least four times per year per crew. Each brigade member must participate in at least two drills per year.

Quantification of fire brigade effectiveness within the fire PSA takes into consideration pertinent factors such as response time, training, equipment availability, and historical fire brigade effectiveness in the nuclear industry. Additional detail concerning manual fire fighting effectiveness is contained in the Sandia Fire Risk Scoping Study (Reference 4.10.8).

4.8.4 Total Environment Equipment Survival

Fire brigade training at WNP-2 includes the subject of post fire ventilation. Portable ventilation equipment is provided for fire brigade use to facilitate ventilation and removal of combustion products. Ventilation is conducted so as to direct smoke to non-critical areas and to the outside atmosphere. The portable ventilation systems are configured so as to prevent ventilation of products of combustion to rooms containing safe shutdown equipment. The inadvertent migration of small quantities of smoke to areas containing safe shutdown equipment would be of limited quantity and duration. Adverse impact on equipment operability for the expected concentrations of combustion products for the short duration of the potential fire is not considered credible.

The potential for spurious or inadvertent fire suppression system actuation was dealt with in detail in the Supply System's evaluation of IE Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment." Although some potential interactions were identified, in no situations does a scenario exist in which all redundant and/or diverse systems are affected. In addition, the seismic/fire interaction assessment specifically addresses the potential for spurious actuation of fire suppression systems as a result of seismic events. No major concerns were identified.

Operator effectiveness and survival has also been thoroughly considered. Each fire area has a pre-fire plan associated with it. Operators are trained to be knowledgeable on actions for each fire area. At WNP-2, the only fire area in which manual operator actions are necessary to shutdown the plant is the main control room. All other fire areas have one train of safe shutdown equipment to perform safe shutdown.

For the main control room fire scenario, operators are trained to a WNP-2 plant specific abnormal procedure for control room evacuation and plant cooldown from the Remote Shutdown Room. This procedure stipulates which manual actions are necessary for this fire area. Because the operators evacuate the control room in this scenario, all actions occur outside the control room and operator egress paths do not go through areas in which fire or combustion products will be present. Lighting for manual actions has been installed to provide adequate 8 hour lighting. Eight hour battery pack units have been installed in the following areas:

Access Route to Remote Shutdown Room
Control Room
Remote Shutdown Room
Alternate Remote Shutdown Room
Vital 4160V Switchgear Room SM-8
Battery Charger Room No. 2
Division 2 Diesel Generator Room
Access Route from Remote Shutdown Room to Division 2 Diesel Room

The lighting listed above ensures that operators will have a lighted path to all areas of the plant for which manual operator actions must occur for a control room fire scenario.

Additional detail concerning equipment survival is contained in the supporting document, Sandia Fire Risk Scoping Study Evaluation.

4.8.5 Control Systems Interactions

For a control room fire and evacuation scenario, WNP-2 implements the abnormal procedure mentioned in the previous subsection. In the event the control room must be evacuated because of smoke, fire or loss of safe shutdown control circuitry, the reactor will be cooled down from the Remote Shutdown Panel and/or Alternate Remote Shutdown Panel. Division 2 equipment can be controlled from the Remote Shutdown Panel and is protected from the effects of a control room fire.

For a control room fire scenario one operator will manually scram the reactor and then the control room is evacuated. All remaining actions are controlled from the Remote Shutdown Panel. When the Remote Shutdown Panel is manned, transfer switches are actuated to the emergency position to allow assessment of reactor pressure vessel conditions and to separate safe shutdown equipment control circuits which may be exposed to the control room fire from circuits which are necessary for operation at the Remote Shutdown Panel. Similar switches are located at the Alternate Remote

Shutdown Panel, DG-2 local control panel, and Division 2 switchgear cubicles. Positioning of these switches to the emergency position allows complete isolation from the control room of electrical circuits necessary for safe shutdown and allows operators to perform shutdown and cooldown from outside of the control room.

4.8.6 Adequacy of Analytical Codes

The WNP-2 IPEEE Fire PSA uses the COMPBRN IIIe code (Reference 4.10.1) which has been modified to account for deficiencies which initially raised the issue of inadequate quality of a fire modelling code. COMPBRN IIIe has been accepted by the NRC, thus, no further effort is required for this issue.

4.9 Sensitivity Analysis

The sensitivity studies included in this section are provided to illustrate the importance and to provide insights of several factors and assumptions used in the Fire PSA. Because the refinements to the initial screening analysis were used to reduce the conservatisms inherent in the screening approach, these refinement assumptions will be examined closely for their importance to the CDF value. No attempt was made to reduce the Control Room Fire contribution to CDF, hence, the sensitivity analyses presented below are applicable to the other fire areas.

The CDF for the Fire PSA was determined for each fire area as the product of a) the initiating event frequency, b) the manual or auto-suppression failure factor, and c) the conditional core damage probability (i.e., given a fire occurs, the probability the failed equipment results in core damage). Therefore, the sensitivity and importance analysis is presented below in terms of each of these three factors.

4.9.1 Initiating Event Frequency Sensitivity

Several more realistic adjustments were made to the initiating event frequencies for the dominant fire areas. These refinements included such assumptions as: a) the probability of transient combustibles present in significant quantity during welding (0.1 factor used), b) barrels of oil are not permanently stored in the dominant fire areas and a time of presence was used (240 hours present), c) in the electrical equipment rooms, several small totally enclosed cabinets could not be ignition sources, and d) small fires, such as pump fire, has small likelihood of propagating to damage cables (0.1 factor used). Not all of the factors were applicable to all of the areas. If all of the adjustments to the initiating frequency of the dominant fire areas were removed, that is, returned to the initial screening value, the Fire CDF would increase by 21%.

An assumption was made that for cable initiated fires in trays of qualified cabling, there was a 95% expectation that the cables would self-extinguish, i.e., a probability of 0.05 the fire would propagate. This assumption was only applied to RC-2 Cable Spreading Room, RC-3 Cable Chase, and TG-1K (portion) Turbine Generator Building Corridor. A sensitivity study was performed by eliminating this factor and resulted in an increase in CDF of 13%.

4.9.2 Auto/Manual Suppression Sensitivity

The automatic detection and suppression capability was credited for the diesel generator areas, the cable spreading room, and the cable chase fire areas. No credit was taken for manual fire suppression action in any fire area. The sensitivity analysis was performed first examining the effect of taking no credit for automatic detect and suppress features. The only areas where the automatic suppression is of consequence is DG-2, DG-3, RC-2 (cable spreading room) and RC-3 (cable chase). The other fire areas remain less than the screening criteria of 1E-06 even without automatic suppression. However, for the four areas noted that do not screen out, the automatic suppression feature is very important. The sensitivity analysis shows CDF increasing 1165% if there was no automatic suppression in DG-2, DG-3, RC-2 and RC-3. An increase in unavailability of this feature by a factor of three, i.e., from 0.025 to 0.075 results in a CDF increase of approximately 75%.

The analysis did not take credit for the suppression effect of having a fire watch posted during welding operations. WNP-2 procedures require an individual present during welding to specifically watch for welding initiated fires and to prevent propagation if one should start. The importance of this conservatism was analyzed and would reduce the CDF by 43% if credited. This is consistent with manual suppression factors which also were not credited in the analysis.

4.9.3 Recovery Factor Sensitivities

The recovery factors discussed in Section 4.6 were used to reduce the Conditional Core Damage Probability (CCDP). The principle factors included a) recovery of the safety-related AC power buses which in turn allowed reopening the MSIVs to use the condenser or to regain use of the containment vent, and b) limiting transient combustibles in the Div 1 and Div 2 Electrical Equipment Room. The CCDP recovery factors reduced the CDF by 81%. As discussed in Section 4.6.5, the analysis did not take credit for the ability to repair or to recover from maintenance for the equipment not lost because of the fire. It is estimated, using WASH-1400 mean time to repair values, that the CDF could be reduced by 20-30% accounting for this factor.

4.10 References

4.10.1 University at Los Angeles, Los Angeles, California, COMPBRN IIIe: An Interactive Computer Code for Fire Risk Analysis; EPRI NP-7282, Project 3000-39, Final Report, May 1991

4.10.2 FIVE Methodology Plant Screening Guide, Professional Loss Control, Palo Alto, CA: EPRI, April 1992, EPRI TR-100370

4.10.3 WPPSS-FTS-133, IPE, July 1994, Rev. 1

4.10.4 WNP-2 FSAR, Appendix F, Fire Protection Evaluation, Amendment 45, July 1992

4.10.5 Report on Full-Scale Horizontal Cable Tray Fire Tests, FY1988, FNAL-TM-1549, September 1988

4.10.6 WNP-2 Cable And Raceway Penetration Schedule (CARPS) [Consists of P&IDs: E550 - Power Cables, E551 - I&C Cables, E550-1 - Power Raceway, E551-1 - I&C Raceway]

4.10.7 Appendix III, Table III-5-3, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, 1975

4.10.8 NUREG/CR-5088, Fire Risk Scoping Study, Sandia National Laboratory, 1

4.10.9 NSAC-181, Fire PSA Requalification Studies, March 1993

4.10.10 EPRI Fire Test Summaries Database, EPRI NP-6343-L, June 1989

5.0 HIGH WINDS, FLOODS, AND OTHER EVENTS

This evaluation is part of the response to the NRC's Generic Letter 88-20, Supplement 4 (June 28, 1991) which requested that all utilities perform a systematic IPEEE to identify any vulnerability to severe accidents. The selection of an external event for this evaluation depends on its frequency of occurrence, magnitude, proximity, and consequences. The external events considered for the WNP-2 site are listed in Section 1.3. Some of those events do not pose a significant threat of a severe accident. Most have been considered during the design stage and contribute insignificantly to the overall plant risk. Others are studied under different programs, such as the Individual Plant Examination (IPE) where lightning and severe cold weather conditions are included (based on operating history) in the initiating frequency calculation for the Loss of Offsite Power (LOOP) initiating event. These are discussed in more detail in the following sections.

The five major external events considered as part of the IPEEE program (Generic Letter (GL) 88-20, Supplement 4) are:

1. Seismic Events

2. Internal Fires

3. High Winds and Tornadoes

4. External Floods (high water, high precipitation, dam failures, and combinations of high rains and dam failures)

5. Transportation and Nearby Facility Accidents (aircraft crashes on the power plant site, ship/barge collisions with power plant structures and ship/barge, truck, railroad, gas/oil/chemical pipeline accidents near the power plant site which release hazardous materials, and facility accidents near the power plant site which release hazardous materials).

The last three items on the list, and any site specific hazards (such as volcanoes) are the Other External Events. Other events that were considered for WNP-2 include (Reference 5.6.1):

1. Extreme Heat,
2. Extreme Cold,
3. Ice,
4. Hail,
5. Snowstorms,
6. Dust storms, Sandstorms,
7. Lightning Strikes,
8. External Fires (i.e., forest fires, grass fires),
9. Extraterrestrial Activity (i.e., Meteorite Strikes, Satellites),
10. Volcanic Activity,
11. Damage or Destruction due to Military Action,
12. Avalanche, Landslide,

13. Release of Hazardous Materials from On-site Storage, and
14. Accidents from Nearby Industrial or Military Facilities.

The progressive screening approach recommended in GL 88-20 was followed in this evaluation to assess these events. This approach is used in screening insignificant or non-applicable external events (see Figure 5-1). One or more of the steps shown in Figure 5-1 can be bypassed if vulnerabilities are either identified or proved to be insignificant. In order to effectively apply the screening approach, some engineering judgment is necessary. Consistent with engineering practice, expert opinions, simplified scoping studies, and bounding analyses were used in forming these judgments. For example, some of the events can be screened out based on initiating frequency (hazard) combined with some bounding evaluation of plant damage and consequences that may indicate that the external event risk is comparably lower (e.g., 1/100) than those of other external or internal events (provided they are less than 1E-6/yr) and can be eliminated from further examination. The NRC suggested screening criteria that were followed for WNP-2 are listed below (an event can be screened out if it meets one of these criteria):

1. The event under consideration is of less or equal damage potential than those for which the plant is designed. To determine this, the plant design bases should be reviewed to assess the resistance capability of the plant to the particular external event. For example, safety related structures are designed for earthquake and tornado loading and can safely withstand a 1-psi peak positive incident overpressure from explosions. Thus, external events producing less than 1 psi need not be considered (as long as it is confirmed that the structures were in fact designed to the specification).

2. The external event has a significantly lower ($\sim 1/10$) mean frequency of occurrence than other events with similar uncertainties and the consequence is equal to or lower than other events.

3. The event is likely (or known) to occur far enough away or at low enough magnitude for a given distance as to not to affect the plant. Examples may include landslides, volcanic eruptions, earthquake-fault ruptures, and explosions.

4. The event is included in or covered by the definition of another event.

   For example, in the event of an aircraft or turbine missile impact on the safety related structures, if the event has the potential to cause core damage but the occurrence frequency is much lower than other core damage frequencies (provided core damage frequency is sufficiently low, e.g. on the order of 1E-6 per year or less), then the events can be screened from detailed evaluation.

Since WNP-2 is a relatively newer plant, the 1981 version of the Standard Review Plan (SRP) was used by the NRC (in reviewing the design for Operating License (OL)), and by the Supply System to assess the safety design margin of the plant. Therefore, the 1981 version of the SRP was used in this evaluation, rather than the 1975 version as permitted by the Generic Letter*[1]. If the WNP-2 design pertaining to an external event satisfies the SRP, and no significant change to the design or the severity of the external event (expected at the site) has occurred since the issuance of the OL, then the contribution from that hazard to the overall core damage frequency can be expected to be less than 10^{-6} per year, and the event can be screened from further evaluation (see GL 88-20, Supplement 4).

5.1 Volcanic Activity

Mountain peaks that are recorded to have erupted within historic time are considered active volcanoes. All of the active volcanoes in the continental U.S. are located in the Cascade mountain range along the Pacific coast. The major volcanoes in the Cascade Range west of the WNP-2 site are Mount Adams (approximately 165 km) and Mount St. Helens (approximately 220 km) west-southwest of the site (FSAR, Section 2.5.1.2.6.1). Mount St. Helens is the most active volcano in the continental U.S., and erupted most recently on May 18, June 12, July 22, August 7 and October 16-18 of 1980. It is located 88 km northeast of Portland, Oregon, 137 km south of Seattle, Washington and 224 km west of Richland, Washington (Note: WNP-2 is 19km northeast of Richland). It is one of 14 volcanoes located in the Cascade mountains between Lassen Peak in northern California, and Mt. Garibaldi east of Vancouver, B.C., Canada. At the time of first eruption (the largest of the series of eruptions in 1980), WNP-2 was still under construction and the nearest nuclear plant, Trojan, was shut down for refueling. Effects of the eruption at the Trojan site included debris in the Columbia River which threatened the ultimate heat sink water supply, and volcanic ash in the air which forced the Trojan plant to isolate its ventilation system and go into the recirculation mode. In addition, the roof top loading was a concern, particularly for nonsafety grade structures. However, potential roof top over loading was alleviated by plant personnel clearing the ash fall from the roofs of the plant structures.

Although major processes and secondary effects of an erupted volcano can be numerous, world-wide data regarding volcanic eruptions and processes show that, except for ash fall, the major volcanic processes (hazards) generally occur within about 40 km of an explosive volcano (FSAR, Section 2.5.1.2.6.1). Because WNP-2 is 165 km east of the closest Cascade composite volcano (Mount Adams), and since the site is not downstream on a drainage emanating from a Cascade composite volcano, only ash fall poses a hazard to the WNP-2 site. Mud flows, avalanches, pyroclastic rock flows, lava flows, and shock waves that may be associated with volcanic activity are confined to the immediate area of the volcano and do not pose a hazard to the WNP-2 site.

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* Note that the Generic Letter permits comparison with the 1975 SRP. Since the WNP-2 Safety Evaluation Report (SER) used the 1981 SRP, it is also used as the reference in this evaluation. Therefore, all SRP sections listed in this document refer to the 1981 edition of the SRP unless otherwise stated.

After the May 18, 1980 eruption of Mount St. Helens, it was estimated that two cubic kilometers (2.0E9 m^3 or 7.0E10 ft^3) of ash and rock were ejected during the violent eruption. Approximately 1300 feet of the 9677 feet peak were blown away, leaving a huge horseshoe crater on its northern flank. The first warning of this event was a minor venting of steam and ash on March 27, 1980. The principal factor in the dispersal of ash is the vertical coherency in the direction and velocity of the high altitude winds. Fairly steady state wind conditions prevailed during the May 18, 1980 eruption of Mount St. Helens, and ash began falling at the WNP-2 site between 2 and 3 hours after the eruption.

The Supply System has completed comprehensive analyses of potential hazards associated with maximum probable ash fall at the site following a volcanic eruption (FSAR Section 2.5.1.2.6.1) and submitted the findings to resolve the issue with the NRC (FSAR Question 360.009). Several areas of concern are specifically addressed in the FSAR and the corresponding internal technical memorandum. The NRC review of the volcanic hazard analysis at WNP-2 is reported in Reference 5.6-3, Supplements 1 and 3. After the initial review (Reference 5.6-3, Supplement 1, Appendix G) of the FSAR, the NRC and the United States Geological Survey (USGS) indicated that the FSAR estimates of the uncompacted thickness of ash fall and the rate of fall were not as conservative as more recently developed estimates by the USGS based on measurements from the May 18,1980 Mt. St. Helens eruption would suggest. In accordance with SRP Section 2.5.1, regarding new information on the regional and site geology learned after issuance of OL, the Supply System was requested to assess the impact of new ash fall data on WNP-2. As a result of this request, the Supply System established a task force to evaluate and recommend a plan to incorporate the findings.

The task force results were reported to the NRC in a letter dated October 4, 1982 (Reference 5.6.14). The WNP-2 plant systems and equipment were evaluated with respect to operability and reliability during a design basis ash fall using the new and more conservative values for maximum compacted and uncompacted ash fall thicknesses of 7.5 cm (3 inch) and 18.8 cm (7.4 inch), respectively. The average and maximum ash fall rate of 0.89 cm/hr (0.35 inch/hr) and 1.12 cm/hr (0.44 inch/hr), respectively, were assumed coincident with loss of offsite power for 2 hours. In addition, the Supply System agreed to develop a warning system tied to the USGS warning system. Based on the results of the task force evaluation, a specific plant procedure was developed that utilizes appropriate equipment modifications and designated new filters for safety related systems at nearby trailers to ensure safe operation and shutdown of the plant following a design-basis ash fall.

Design Basis Ash Fall

The following table compares the results of calculated ash values using information from the WNP-2 FSAR, the WNP-2 SER Supplement No. 1, and the May 18, 1980 Mt. St. Helens eruption. Although, the Supply System believes the FSAR values are a more accurate estimate of actual conditions expected during a design basis ash fall, the more conservative values from the SSER, coincident with loss of offsite power for two hours, were used by the task force to evaluate the WNP-2 plant systems and equipment with regard to operability and reliability during design basis ash fall.

Table 5.1-1 Design Basis Ash Fall Parameters

Parameter [Units]	Mt. St. Helens 1980	FSAR	SER
Maximum Ash fall [in] uncompacted	2.75	5.00	7.4
Maximum Ash fall [in] compacted	1.65	3.00	3.00
Ash fall Duration [hrs.]	20	20	20
Ash fall Rate [in/hr] Average	0.14	0.25	0.35
Ash fall Rate [in/hr] Maximum	0.21	0.35	0.44
Average Grain Size [μm]	75	75	75
Density [lb/ft^3] compacted	96	96	96
Compaction [%]	40	40	60
Air Concentration [μgm/m^3] Average	69,795	124,634	174,488
Air Concentration [μgm/m^3] Maximum	104,693	155,793	219,536

The design basis ash fall used in the task force evaluation is equivalent to an eruption that is expected to occur every 2,000 to 3,000 yr (Reference 5.6.15). By comparison, the Mt. St. Helens eruption was a small to moderate volume eruption that is expected to have a return period of 500 to 1,000 years. Since the direction of the axes of the maximum ash fall will not always point to the WNP-2 site, the actual return period of significant ash fall at the WNP-2 site may be much greater than these values.

Effects of Design Basis Ash Fall on WNP-2

Information generated or compiled on the effects of volcanic ash fall at WNP-2 is found in the Task Force Report (Reference 5.6.7), applicable plant procedures, and the WNP-2 Letter to the NRC on this subject (Reference 5.6.14). According to Reference 5.6.7, in a design basis volcanic event concurrent with loss of offsite power during ash fall, the following minimum set of systems will be required for hot and cold shutdown of the reactor, assuming reactor scram by closure of the MSIVs:

Diesel Generator
RCIC
RHR
SW
Associated HVAC Units
ADS

This list is consistent with systems modeled in the LOOP event tree in the WNP-2 IPE. If the offsite power is not lost, the plant will remain operating while both shutdown trains are maintained as ash-free as possible. A decision to shut down is predicated on the amount of ash fall that enters the peripheral buildings, particularly the circulating water pumphouse. The maintenance of an ash-free environment is the key to successfully coping with a volcanic eruption that results in ash fall at the WNP-2 site. In order to ensure an ash-free environment, adequate filtration systems are provided in two SEA storage vans nearby for ventilation/filtration systems associated with the following safe shutdown systems:

- Diesel Generator Building Combustion Air Intake and Ventilation System
- Standby Service Water Pump House(s) Ventilation System
- Reactor Building Emergency Cooling and Critical Electrical Equipment Room Cooling Systems

The task force reported that fluid systems such as the standby service water pond present no major problems, but seals and packing should be maintained during the event. Ash buildup on electrical equipment can cause failure by thermal overloading and by shorting. However, it is not possible to quantitatively predict ash effects on critical electrical equipment due to the number of different types of equipment and the uncertainty involved in predicting the ash thickness. Therefore, on-going preventive maintenance (e.g., sweeping of ash) must be utilized to keep the safety related, safe shutdown electrical equipment from failing during and after the ash fall until offsite power is restored. Instrumentation, both fluid and pneumatic, is not affected by ash.

The task force evaluation has shown that protection of equipment necessary for safe shutdown can be provided with minor modification to some of the HVAC systems together with the initiation of maintenance activities on the standby components in the event of an ash fall to minimize ash buildup. Based on the task force recommendations and subsequent NRC correspondences (including inspection reports that dealt with the subject), the Supply System has implemented an abnormal condition procedure entitled "Design basis ash fallout" (PPM 4.12.4.5, Rev. 10, July, 1994). That PPM contains details on the definition of design basis ash fall at the site, as well as the specific steps taken in response to a volcanic event. Among those steps are notification of Supply System and other personnel for activation of emergency centers, installation of temporary filters, inspection of I&C circuits and cabinets for ash fall buildup, and implementation of sweeping activities to maintain equipment and structures as free from ash fall buildup as possible.

<u>Summary</u>

Based on the task force assessment and the resulting plant procedure modifications including the addition of advanced warning provisions and equipment modifications described above, the Supply System has concluded that WNP-2 is assured of safe plant operation and shutdown following a design basis ash fall. The NRC has concurred with the Supply System and concluded that the Supply System analysis, procedures and equipment modifications will provide adequate assurance of safe plant operation and shutdown following a volcanic event (SSER Section 2.5.1.3, supplement 3 and Reference 5.6-16). Therefore, it is concluded that no further examination of this event will be necessary.

5.2 High Winds/Tornadoes

High winds from tornadoes, hurricanes or wind storms are a potential threat to nuclear power plant operation due to the risk associated with potential damage to safety related equipment resulting from failure of structures. The structures can be damaged due to the pressure differential caused by tornado dynamics, missiles generated by a tornado, or damage induced by high wind loading directly on the structures. The applicable NRC regulations, and WNP-2 plant procedures that deal with protecting safety related systems and structures against potential high winds/tornadoes are listed below:

- 10CFR Part 50 Appendix A General Design Criteria for Nuclear Power Plants, Criterion 2 - Design Bases for Protection against Natural Phenomena.
- 10CFR Part 50 Appendix A GDC, Criterion 4 - Environmental and missile design bases
- 10CFR Part 100.10 Factors to be considered when evaluating sites
- RG 1.13 Fuel storage facility design basis
- RG 1.76 Design basis tornado for nuclear power plants
- RG 1.117 Tornado design classification, structures, systems & components to be protected against tornadoes
- SRP 2.3.1 Regional climatology
- SRP 3.3.1 Wind loading
- SRP 3.3.2 Tornado loading
- SRP 3.5.1.4 Missiles Generated by Natural Phenomena
- SRP 3.5.1.5 Site proximity missiles (except aircraft)
- SRP 3.5.2 Structures, systems, and components to be protected from externally generated missiles
- SRP 3.5.3 Barrier design procedures
- PPM 4.12.8 Tornado/High Winds

Wind speeds generated during hurricanes and other storms are less intense and lower in magnitude than those generated by tornadoes. Thus, plant structures that are designed to satisfy the design criteria for tornadoes will also satisfy the design criteria for those events categorized as "high winds." However, for completeness each of these topics is evaluated for compliance with SRP requirements.

5.2.1 High Winds

SRP Section 3.3.1 requires that the licensee must meet the requirements of GDC 2 associated with structural integrity during design basis wind loading. The design basis wind is specified as a velocity of 100 mph at 30 ft above site grade based on a recurrence period of 100 years. As stated in the WNP-2 FSAR, Section 3.3.1, all structures are designed to withstand this basic wind velocity, including gusts. The WNP-2 design reflects the following SRP requirements:

1. The design basis wind speed is higher than the peak wind speed recorded for the site and includes adequate margins for safety.

2. Effects of various plant conditions, including accident conditions, combined with effects of natural phenomena are considered in the design basis.

3. Safety functions to be performed, if needed, are ensured to be available by adequately protecting those structures that house important safety equipment and components against severe wind loading.

As concluded in the SER by the NRC staff, WNP-2 meets these requirements by complying with ANSI A58.1 and ASCE paper No. 3269 (References 5.6.17 and 5.6.18, respectively). Thus, WNP-2 design meets the requirements of the SRP for high winds.

5.2.2 Tornadoes

Nuclear power plants are regulated by 10CFR50 and 10CFR100 in regards to protection against high winds and tornado hazards. Regulatory Guide 1.76 provides, on a conservatively enveloped regional basis, the expected hazards from tornadoes in a given region in the U.S. As shown in Figure 5-2, there are three such regions varying from most severe (Region I) in the eastern part of the U.S., to least severe (Region III) in the western part of the U.S. WNP-2 is in the Tornado Intensity Region III which has the least severe tornado requirements.

Regulatory Guide 1.117 provides guidance on the plant systems, structures, components, areas, etc., that must be protected against high winds and tornadoes. Other NRC reviewed and approved documents, i.e., ASCE paper 3269 and ANSI A58.1 (References 5.6.17 and 5.6.18, respectively), contain detailed information on how to transform wind velocities into pressure loading on structures. They also provide vertical distributions of pressure loading and gust factors. These specific criteria, along with the relevant SRP criteria in Sections 3.3.1, 3.3.2, and 3.5.2.4 have been established to provide an annual probability of exceedance of design loads of 1.0E-7 for a given site.

As stated in FSAR Sections 2.3.1.2.1.3 and 3.3.2.1, WNP-2 design criteria are for wind speeds of 300 mph rotational and 60 mph translational with a pressure drop of 3 psi occurring at 1.0 psi/sec. Compared to Regulatory Guide 1.76 criteria for Region III, i.e., 190 mph rotational and 50 mph translational with a pressure drop of 1.5 psi occurring at 0.6 psi/sec, the current requirements represent a robust design for WNP-2. In accordance with RG 1.76 Section C.2, a comprehensive analysis was done and provided to the NRC for approval to justify the use of more representative and less conservative design basis tornado loading for the WNP-2 site. The new design criteria proposed by the Supply System requires protection against 157 mph rotational and 35 mph translational velocities with a pressure drop of 0.70 psi occurring at 0.24 psi/sec. If this proposed design criteria is approved by the NRC, only new structures and any future modifications to existing structures will be affected by the new criteria since existing structures already meet the requirements of Region III with significant margin. If and when new structures are constructed at the WNP-2 site, the new criteria may apply. However, for the purposes of this assessment, which applies to structures already in place, the current criteria are relevant. The existing criteria are summarized below:

FSAR criteria and the Region III criteria from the SRP and RG 1.76 are shown in Table 5.2-1 for comparison:

Table 5.2-1 Winds Speed and Missile Spectrum

Parameter	WNP-2 FSAR	SRP/RG 1.76, Region III
Rotational speed [mph]	300	190
Translational speed [mph]	60	50
Maximum wind speed [mph]	360	240
Pressure drop [psi]	3	1.5
Pressure drop rate [psi/sec]	1	0.6
Utility pole missile weight [lbs]	1600	1124
Utility pole missile speed [ft/sec]	241	85
Utility pole missile height [ft]	35	35
Utility pole missile diameter [in]	14	13.5
Steel rod missile weight [lbs]	8	8.8
Steel rod missile speed [ft/sec]	259	27
Steel rod missile height [ft]	3	3
Steel rod missile diameter [in]	1	1
Automobile missile weight [lbs]	--	3990
Automobile missile speed [ft/sec]	--	134

It can be seen from Table 5.2-1 and Figure 5-2 that the existing WNP-2 design basis is closer to Region I requirements than Region III. The same SRP requirements as those described in Section 5.2.1 under "high winds" (items 1 through 3 listed in that section) must be followed, but with the higher wind speeds listed above when assessing compliance against the SRP. At WNP-2, all safety related structures housing or associated with safe shutdown equipment were designed to resist a tornado of 300 mph tangential wind velocity and a 60 mph translational wind velocity with a simultaneous atmospheric pressure drop of 3 psi at a rate of 1 psi/sec. The NRC staff in their review of the design (SER 3.3.2) have concluded that ANSI A58.1 and ASCE paper No. 3269 have been appropriately applied to the design to ensure that safety related systems and components are adequately protected against the most severe tornado expected for the site, and the systems and components will perform their intended safety function if needed.

SRP Section 3.5.1.4 "missiles generated by natural phenomena," requires (via GDC 4) that the same structures, systems, and components that are essential to safety and are protected against natural phenomena (as described above) must be protected against missiles potentially generated by tornadoes. The guide line provided in RG 1.76 and RG 1.117 serves as the acceptance criteria for this design basis. The applicable missiles selected for the WNP-2 site in accordance with RG 1.76 for tornado Region III are shown in Table 5.2-1. The NRC staff reviewed this missile spectrum (SER 3.5.1.4) and concluded that the missiles in the spectrum are representative of the site. In addition, the staff concluded that the missile spectrum selected meets the requirements of GDC 2, GDC 4, RG 1.76, and RG 1.117. Thus, WNP-2's current tornado design criteria meets the requirements of the SRP, and no further evaluation is necessary at this time.

5.3 External Flooding and Probable Maximum Precipitation

The WNP-2 design specific to external flood protection was reviewed against GDC 2 by the NRC staff (SER Section 2.4). The GDC 2 section forms the basis for appropriate regulatory guides and SRP criteria by requiring that all safety related structures, systems and components must withstand the effects of natural phenomena, including floods. Specifically, the design must reflect the following requirements:

1. The design basis flood is the most severe of all the floods that have been recorded for the site plus added appropriate margin for safety.

2. Effects of various plant conditions including accident conditions, combined with effects of external flood are considered in the design.

3. Safety functions to be performed, if needed, are ensured to be available by adequately protecting those structures that house important safety equipment and components against the design basis flood.

The applicable Code of Federal Regulations (CFR), Regulatory Guides (RG), and Standard Review Plan (SRP) sections that deal with protecting safety related systems and structures against potential external flood are listed below:

- 10CFR Part 50 Appendix A GDC, Criterion 2 - Design bases for protection against natural phenomena
- 10CFR Part 100, Site criteria
- 10CFR Part Appendix A, Seismic and Geologic Siting Criteria for Nuclear Power Plants
- RG 1.27 Ultimate heat sink
- RG 1.59 Design Basis Floods for nuclear power plants
- RG 1.102 Flood protection for nuclear power plants
- SRP 2.4.2 Floods
- SRP 2.4.3 Probable maximum flood (PMF) on streams and floods
- SRP 2.4.4 Potential dam failures
- SRP 2.4.10 Flood protection requirements
- SRP 3.4.1 Flood Protection

5.3.1 Site Description

The most predominant hydrologic feature of the WNP-2 site is the Columbia River. The WNP-2 site is located on the Hanford reservation, 10 miles north of Richland and approximately three miles west of the Columbia River at River Mile (RM) 352. Over the past 35 years, the river flow has been regulated by dams and reservoirs. A total of seven dams upstream and four dams downstream of the site are utilized to control the river flow within the United States boundary (FSAR 2.4.1.1). The Grand Coulee dam is the largest and most complex of the dams with 9,402,000 acre-feet of available storage. The river surface water level near the WNP-2 site is primarily controlled by regulation of the 35 million acre-feet capacity of upstream reservoir projects. Limited flow control in the immediate vicinity of the site is provided by the regulation of the nearest upstream hydroelectric projects, Priest Rapids Dam (at RM 397), containing about 45,000 acre-feet of active storage, and Wanapum Dam (at RM 415), containing about 161,000 acre-feet of active storage. WNP-2 draws water from the river for cooling water makeup (maximum capacity 55.7 cfs) and plant service water. Based on USGS data, the approximate river bottom is at elevation 328 feet, while the ground elevation at the plant is at 440 feet above Mean Sea Level (MSL). The variations in the river flow throughout the year can vary from low of 36,000 cfs to as much as 160,000 cfs. High flows from 250,000 cfs to 450,000 cfs have been recorded during spring runoff. The average annual flow is 120,000 cfs.

5.3.2 Columbia River Floods

Evaluated Columbia River floods near the WNP-2 site include the Probable Maximum Flood (PMF) expected to occur based on historical data and hypothetical failures of upstream dams. These postulated Columbia river floods are discussed in the following sections.

5.3.2.1 Probable Maximum Flood

Historically, major floods on the Columbia River Basin result from rapid spring melting over a wide area, augmented by rain or above normal precipitation in May. Additional flooding effects are from water swell due to major Chinook winds which cause rapid area temperature rise. The maximum recorded flood occurred on June 7, 1894; it resulted from a combination of hydrometeorologic conditions, including heavy snowpack and rapid melt plus rainfall (FSAR Section 2.4.2.1). The peak discharge of 740,000 cfs at WNP-2 was estimated for this flood from the high water mark at Wenatchee, Washington. The estimated flood level is approximately 373 ft. MSL which is 68 feet below the plant elevation of 441 ft. MSL. The most recent flood, which occurred in 1948, produced a peak discharge of 690,000 cfs and a 369 feet MSL water level near the site. The current regulation of river flow by dams and reservoirs built since the historic floods would significantly reduce the peak discharge if the floods were to occur now. Based on available historical data and analysis by the U.S. Army Corps of Engineers, the Supply System estimates that the probable maximum flood at the WNP-2 site would produce a peak discharge of 1,440,000 cfs and a flood level of 390 ft MSL near the site (FSAR Section 2.4.3). The analysis and the estimate of the PMF value is consistent with Regulatory Guide 1.59, Rev. 2. and the requirements of the GDC 2. Therefore, as concluded by the NRC staff (SER 2.4.3.2.1), the analyses meets the requirements of the SRP.

5.3.2.2 External Floods Caused by Dam Failures

The Supply System has identified worst flood conditions at the WNP-2 site by assuming arbitrary failure of key dam structures. The worst flood was estimated using existing analytical studies of artificial flood waves under various assumptions of catastrophic failure of dams. Based on a previously classified study by the Seattle District Corps of U.S. Army Engineers (FSAR 2.4.4, Reference 26), a postulated failure of the Grand Coulee Dam ("Artificial Flood No. 1" due to enemy attack) would initiate a catastrophic flood that could have a peak discharge of 8,800,000 cfs at the dam and 4,800,000 cfs at river mile 338 near the WNP-2 site. This is the limiting case flood for the Columbia river with a resulting flood level of 424 feet MSL near the WNP-2 site. Failure of any of the other dams would cause a lesser flood because of the considerably smaller size of those dams or natural channel restrictions. As concluded by the NRC staff in the SER (Section 2.4.3.2.2), except for the river intake structure, this conservative flood level is well below the floors of the reactor building structures which are at 441 feet MSL. The river intake structure, however, is not necessary for plant shutdown and cooldown in this case since the seismic category I spray pond, at a finished grade of 434 ft MSL, contains sufficient cooling water to shut the plant down and maintain it in shutdown mode for 30 days. Based on the foregoing information, the NRC staff concluded that the plant design reflects the requirements of GDC 2 with respect to dam failure floods on the Columbia River, and therefore meets the SRP requirements for this event.

5.3.2.3 Other Studies

More recently published results of a study sponsored by the DOE for its N-Reactor (at river mile 379.5) were reviewed to compare with the WNP-2 design basis analysis results. The results for DOE sites are contained in a document produced by Lawrence Livermore National Laboratory, which summarizes more detailed studies published by the same laboratory. The PMF for the N-Reactor site is estimated to produce a peak discharge of approximately 1,200,000 cfs and 1,550,000 cfs for regulated and unregulated discharge, respectively. The corresponding peak water levels at the N-Reactor site are estimated at 418 ft MSL and 425 ft MSL, respectively. This is consistent with the PMF estimates for the WNP-2 site which is approximately 28 river miles downstream (see Section 3.3.2.1).

As for dam failures, the DOE study apparently uses "Artificial Flood No. 2" with a peak release of 21,000,000 cfs at Grand Coulee Dam and a resulting peak discharge of 8,000,000 cfs at the N-Reactor site. The effects of the dam failure are expected to take 20 hours to propagate to the N-Reactor site, which is nearly 220 river miles from the Grand Coulee dam. The DOE and the WNP-2 FSAR studies were both based on the same reference studies by the U.S. Army Corps of Engineers. The Corps of Engineers' studies postulated various breaching scenarios resulting from hypothetical enemy attacks on the Grand Coulee Dam. Although these studies bear no relationship to flooding from natural causes, they were used by the Supply System and DOE as a very conservative limiting case. The peak discharge rate assumed by the DOE study near the N-Reactor site results in flood levels ranging from approximately 412 ft MSL to 450 ft MSL with corresponding frequencies of exceedance ranging from $7x10^{-4}$ to $7x10^{-5}$/year, respectively. The corresponding flood levels at the WNP-2 site would range from roughly 362 ft MSL to 400 ft MSL. This indicates that the flood level conservatively estimated in the FSAR, for a hypothetical Grand Coulee dam failure, is conservative even if one were to postulate the artificial flood number 2 event which was postulated for national security purposes by the U.S. Army Corps of Engineers. Therefore, based on the foregoing discussion, no further evaluation of this event is necessary.

5.3.3 Probable Maximum Precipitation (PMP)

The Probable Maximum Precipitation (PMP) for WNP-2 is estimated at 9.2 inches in a six hour period based on available U.S. Weather Bureau estimates (FSAR Section 2.4.3.1). The corresponding maximum flood water level at the spray ponds, including the effects of the wind setup and wave runup, is 433.3 ft MSL, which is less than the spray pond finished wall elevation of 435.0 ft MSL. This PMP value can be considered extremely conservative when compared to the maximum recorded precipitation of 1.68 inches in six hours occurred in October 1957. The greatest recently recorded snow depth of 15 inches was observed in January 1993 (FSAR Section 1.2.2.1.2.4). The

PMP value is higher than site specific historic rainfall would suggest because it is based on readings from two reference weather stations that are located on the west side of the Cascade mountain range. These stations are included in the Pacific Northwest data in the Weather Bureau's report. However, the eastern part of the Cascades, where WNP-2 is located, is in the rain shadow of the mountains and receives significantly lower precipitation. The National Weather Service, the agency that now produces the hydrometeorological reports, was contacted to obtain an updated report (the current report No. 43 was issued in 1966) per NUREG-1407 requirement to address Generic Issue 103.

According to the FSAR (Section 2.4.2.3), the roofs of the buildings are designed with adequate drainage to withstand the PMP. A storm sewer system carries the discharge from the roof to a manhole located southeast of the Reactor Building. The manhole is connected to a pipeline which transports the water to a low elevation point 1,500 feet northeast of the plant. Even if the drainage system is completely blocked, overflow scuppers limit the depth of water to within the design load carrying capability of the roofs. Those safety-related structures that do not have this relief capability can structurally carry the entire PMP accumulations. In this case the structural distress level is higher than the corresponding parapet height. Details of these findings are provided in the Supply System response to the NRC question 371.015. The analysis included the diesel generator building, radwaste/control building and adjacent corridor, the reactor building and adjacent corridor, and the standby service water pump house. The PMP value used for the analysis was the 11.7"/6 hr point value reported in an earlier Amendment of the FSAR. Since then the PMP numbers have been revised to 9.2"/6 hr which is 21% less than the value used in the analysis. As discussed above, even the revised PMP values are very conservative when compared to the precipitation that can be expected based on historical data in the rain shadow region, east of the Cascades, including the WNP-2 site. Therefore, the higher value used by the Supply System in the roof loading analysis is still representative of the revised PMP values.

Based on the foregoing information, including conclusions by the NRC staff in the WNP-2 SER, the roofs of safety-related buildings, the diesel generator building, radwaste/control building and adjacent corridor, reactor building and adjacent corridor, and the standby service water pump house are adequately designed for the PMP and meet the requirements of GDC 2. Thus, the WNP-2 plant design meets the requirements of appropriate sections of the SRP pertaining to this external event and therefore, can be screened out from further evaluation.

5.4 Transportation accidents and nearby facility accidents

This external event is concerned with potential accidents due to activities other than those at the WNP-2 site for the purpose of power production. These are hazards that are postulated to result from accidents during nearby (within 5 miles) transportation, industrial, and military activities, including accidental release from on-site hazardous material storage. Transportation activities which must be considered include aviation, marine (ship/barge), pipeline (gas/oil), railroad, and trucks carrying hazardous materials. All of these potential activities near the reactor site were extensively analyzed during the plant siting process and are documented in the FSAR (Section 2.2, entitled "Nearby Industrial, Transportation, and Military Facilities"). The WNP-2 location reflects the requirements of the regulatory documents listed below:

- 10CFR Part 50.34 Contents of Applications; Technical Information,
- 10CFR Part 100 Reactor site criteria
- 10CFR Part 100.10 Factors to be considered when evaluating sites
- RG 1.78 Assumptions for evaluating habitability of a nuclear power plant control room during a postulated hazardous chemical release
- RG 1.91 Evaluations of explosions postulated to occur on transportation routes near nuclear plants
- RG 1.95 protection on nuclear power plant control room operators against an accidental chlorine release
- SRP 2.2.1 - 2.2.2 Identification of potential hazards in the site vicinity
- SRP 2.2.3 Evaluation of potential accidents
- SRP 3.5.1.5 Site proximity missiles (except aircraft)
- SRP 3.5.1.6 Aircraft hazards

The NRC staff reviewed the WNP-2 plant in accordance with SRP Sections 2.2.1, 2.2.2, 2.2.3, 3.5.1.5, and 3.5.1.6 (SER Section 2.2). Within a 5 miles radius of the WNP-2 site, there are no military bases, missile sites, manufacturing plants, chemical plants, chemical storage facilities or airports. Included within the 5 miles radius are the Supply System Plant Engineering Center, the H.J. Ashe Substation, the DOE's Fast Flux Test Facility (FFTF), the WNP-1 and 4 construction sites, the Wye radioactive waste burial ground and a permanent meteorological tower (FSAR Section 2.2.1, Amendment No. 46, August 1992). The following sections provide additional detail.

5.4.1 Transportation Accidents due to Aircraft Activity

SRP Section 3.5.1.6 provides the required criteria for siting nuclear plants near airports and/or airways. If the distance at which aircraft activity occurs meets the requirements listed below, the probability of an aircraft accident resulting in radiological consequences greater than the 10CFR Part 100 exposures guideline can be considered to be less than 10^{-7} per year.

1. The plant-to-airport distance D is between 5 and 10 statute miles and the projected annual number of operations is less than 500 D^2, or D is greater than 10 statute miles and the number of operations is less than 1000 D^2;

2. The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation;

3. The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

There are no airports within 10 miles of the WNP-2 site, but there are three commercial airports and four private airports within twenty miles of the site. Details of the airports, including the flight frequency and distance from the WNP-2 site, are provided in Table 5.4-1 (FSAR Section 2.2.2.5). The airspace over the Hanford site is periodically used as a marshaling area for military aircraft participating in training missions on the Yakima Firing Range. Such operations are scheduled at two to three year intervals with up to six aircraft per day passing over the site for eight days of exercises. All operations are conducted under visual flight rules conditions at more than 1000 feet above ground level.

Table 5.4-1 Airports Nearby the WNP-2 Site*

Airport	Distance (D) From Plant	Flight Per Year	1000 D^2	Orientation From Site	Type of Operation
Richland	11 miles	40,000	1,210,000	South	Commercial
Tri-Cities	17 miles	75,659	2,890,000	Southeast	60% General Aviation 39% Commercial 1% Military
Vista	18 miles	~20,000 (50 -60 per day)	3,240,000	South Southeast	
McWhorter	18 miles		3,240,000	Southwest	Private
Kent Farms	12 miles		1,440,000	Southeast	Private
Hathaway Ranch	14 miles		1,960,000	Northeast	Private
Hanford	15 miles		2,225,000	Northwest	Closed since 1976

* FSAR values reported here as of August, 1993. In August 1994 these values were updated and are described in the text.

As indicated in the table, there are no airports located within 10 miles of the site, and airports located greater than 10 miles from the site do not have operations greater than $1000D^2$. Figure 5-3 (same as FSAR Figure 2.2-3) shows the airports, low level federal airways, and airport instrument approaches in the vicinity of the WNP-2 site. The federal airway route information listed in the FSAR (Section 2.2.1) was updated in FSAR Amendments 46 and 48, on August 1992 and 1993, respectively (associated FSAR Section 3.5.1.6 was not updated).

In August 1994, Richland, Tri-Cities, Vista, and the Yakima army air fields were contacted by telephone to inquire about any changes since the last update. The fixed based operator at Richland airport indicated that the values for existing number of flights per year reported in the FSAR (40,000) seem to be an order of magnitude larger than what he has been observing. According to the airport operator, 10 flights per day on average can be expected. If landing and take off occur out of the Richland airport for those flights, the total flights per year would be less than 8,000, but were realistically estimated at approximately 4,000 flights per year. (Note: there are currently 75 aircraft based at the Richland airport.)

For the Tri-Cities airport, the manager's office and the FAA control tower were contacted. The number of flights and aircraft associated with the airport were not updated. The FAA provided a conservative estimate of 5840 flights per year that can be expected through airway V187. Airway V187 is the only airway that can pose a potential threat to the WNP-2 since it is within 2 miles of the plant (Figure 5-3). The number obtained for airway V187 is for flights by aircraft that file Instrument Flight Rule (IFR) flight plans. Aircraft that fly Visual Flight Rules (VFR) do not have to file a flight plan, and as a result, records of those flights are not saved. For the purpose of this evaluation, the total aircraft flying through airway V187 is conservatively assumed to be 10,000 per year. Other flight paths shown on Figure 5-3 are either sufficiently far away (more than two miles) with sufficiently low volume or are not currently active. According to the airport manager, the 12 NM and 14 NM arc shown on the figure is no longer used, except on a rare occasion.

For the Vista airport in Kennewick, the airport owner said that the number of flights and aircraft associated with that airport has not changed since the last FSAR update. Lastly, Yakima army airfield aviation manager was contacted to obtain updated information regarding military airspace usage. He said that 6 aircraft per day on an 8 day training mission every two to three years is still the current volume. None of the private airports were contacted since they are more than 10 miles away and would not have sufficient flight volume to pose any threat to WNP-2.

Based on the foregoing discussion and the information provided in Table 5.4-1, only the Federal Airway V187 (located over WNP-2 with a minimum altitude of 3500 feet MSL), could pose a potential threat to the WNP-2 plant. Since this is a federal airway that passes through the vicinity of the site, the probability per year (P_{FA}) of an aircraft crashing into the plant is given by NUREG-4052 and SRP 3.5.1.6:

$$P_{FA} = C*N*A/W$$

where;

C = In-flight crash rate per mile for aircraft using airway (4E-10/mile, SRP 3.5.1.6)
N = Number of flights per year along the airway (10,000, from Tri-Cities airport data
W = Width of airway in miles (3.8 miles, FSAR 3.5.1.6.1, Reference 35)
A = Effective area of the plant in square miles (0.02 sq. mile, Safety related buildings roof area plus 25%, FSAR vol. 25, response to Q.371.015)

Therefore, the probability per year is estimated as 2.0E-8, which is sufficiently low as to screen this airway from further evaluation.

As concluded by the NRC staff, and illustrated in this report, most aircraft activities are sufficiently distant from the WNP-2 site as to not to pose a threat to safe operation of the plant. Those activities that come near the site or pass over the site occur infrequently and at sufficiently high elevations that they are unlikely to pose a threat to the safe operation of the plant. Therefore, this event is screened from further evaluation.

5.4.2 Transportation Accidents Due to Marine (Ship/Barge) Activity

Because WNP-2 is located near the Columbia river, potential hazards due to river craft accidents were assessed during the siting stage of the plant design (FSAR 2.2.1). Some river barge traffic exists as far upstream as the Ports of Pasco and Kennewick and the Port of Benton docking facility in north Richland. Traffic to the north Richland dock is not as frequent as that in Pasco and Kennewick due to little industrial activity in the region between Richland and Priest Rapids Dam. Near the WNP-2 site, there is no commercial river traffic passing the site; some small pleasure boats may occasionally pass the site (FSAR Section 2.2.3.1). Therefore, based on the foregoing discussion this event is not evaluated any further.

5.4.3 Transportation Accidents Due to Pipeline Activity

There are no oil or gas pipelines passing near the WNP-2 site. Within 30 miles of the site there are two natural gas pipe lines at 12 miles and 24 miles away from the site. The nearest petroleum product storage tanks are at the marine terminal at Big Pasco (13.5 million gal), the Chevron Pipeline Company (25 million gal), and at the Tidewater Barge line (23 million gal), all located 22 miles southeast of WNP-2 (FSAR 2.2.2.3 and SER 2.2.2). These facilities are at a sufficient distance from the site such that there are no potential hazards to safe operation of the plant due to a natural gas fire or explosion.

5.4.4 Transportation Accidents Due to Railroad Activity

The DOE-owned and operated mainline railroad track is used in support of the Hanford operation. At its nearest point, the railroad is 510 feet from the reactor building (Figure 5-4). Large quantities of hazardous materials were shipped on this track during the initial licensing stage of WNP-2, but such shipments are no longer made (FSAR 2.2.2.2, Amendment No. 48, August 1993: a note from Westinghouse Hanford). Although no shipments of explosives of more than 1,800 pounds are expected to be made, the DOE Richland operations office has agreed to notify the Supply System prior to transporting any explosives of such magnitude past the WNP-2 site. If a shipment is to be made, the Supply System will provide an analysis to the NRC of the potential consequences prior to the start of such a shipment (FSAR 2.2.2.2, Amendment No. 48, August 1993). As described in the FSAR (Section 2.2.3.1), the Supply System has investigated resistance of plant structures to explosions. Based on the design basis requirement for safety-related structures to withstand the worst possible combination of wind velocity and associated pressure drop due to a design bases tornado, it has been determined that the reactor building can resist an explosion of 20,000 lbs of dynamite on a railway car 510 feet from the reactor building (FSAR 2.2.3.1). Based on the above information, the NRC staff concluded that the railroad will pose no undue hazard to the safe operations of WNP-2 (SER Section 2.2.1). Therefore, this event is screened from further evaluation.

5.4.5 Transportation Accidents Due to Truck Activity

The WNP-2 site is serviced by a two-lane paved access road connected to the DOE road system (FSAR 2.2.1). State Highway 240 traverses the Hanford reservation from the southeast to the northwest and passes within about 7 miles of WNP-2 in the southwest quadrant, as can be seen from Figure 5-5 (FSAR Figure 2.2.1). From Regulatory Guide 1.91, explosive blasts generating 1 psi or less in overpressure on site buildings due to a shock wave can be dismissed from further consideration, if it can be verified that the site buildings of concern can withstand a 1 psi overpressure. According to the Regulatory Guide, a 1500-lb truckload of TNT at a distance of 510 feet (equivalent to the separation between the DOE railroad tracks and the Reactor Building) will generate such a shock wave. As discussed in Section 2.2.3.1 of the FSAR, the reactor building of WNP-2 can resist an explosion of 20,000 lbs. of dynamite on a rail car 510 feet from the reactor building. Furthermore, all road ways passing the site are much further away than 2000 ft. Thus, nearby truck activity does not pose undue risk to the plant and this event need not be evaluated any further.

5.4.6 Nearby Industrial Facilities

Because of potential threat due to explosions at a nearby industrial facility, causing hazardous material to release and generating missiles, nearby industrial activities were reviewed during the site evaluation for the WNP-2 Plant (FSAR 2.2). The effects of an explosion from these facilities, if it were to occur, would be similar to those postulated due to nearby railroad and trucking activity. As a first step in evaluating the potential for this event, facilities, if any, that are within 5 miles of the reactor site must be identified and their activity examined for potential threat. Near the WNP-2 site, the following facilities are within the 5 mile radius:

1. WPPSS Engineering Center
2. H.J. Ashe substation
3. DOE Fast Flux Test Facility (FFTF)
4. WNP-1/4 sites
5. Wye 300 north radioactive waste burial ground
6. Permanent meteorological tower.

The plant engineering center is located west of the WNP-2 turbine generator building. It is a two-story 100,000 square foot facility designed to house 470 WNP-2 plant staff personnel. The H.J. Ashe substation is part of the BPA transmission system, and is located approximately 1/2 mile north of WNP-2. The permanent meteorological tower is located less than 1/2 miles west of the plant site. Less than 6000 feet from the WNP-2 site are the WNP-1 and 4 sites; construction activities at these plants have been permanently suspended. The Wye Burial ground for radioactive material is located immediately west of the WNP-2 plant site. Another waste disposal site, 300 North, is located about 3-1/2 miles south of the Supply System projects. None of these facilities contain activity or materials that will cause undue risk to WNP-2 plant operation (FSAR 2.2).

Lastly, the FFTF is 3 miles southwest of the WNP-2 site. It is a sodium cooled fast reactor used for testing reactor fuel elements. A potential hazard from sodium oxide release from FFTF is presented in the FSAR. In the FSAR (Section 6.4.4.2.2), a conservative analysis is presented that considers the dispersion characteristics of sodium oxide smoke resulting from a postulated loss-of-coolant accident at the FFTF. The analysis used appropriate Regulatory Guide 1.78 information and came up with the worst-case wind speed (1.2 m/s) that results in the highest concentration (8.7 mg/m^3). This allows FFTF operators 55 minutes to notify the WNP-2 control room as required by the FFTF emergency plan. This time is adequate for WNP-2 operators to mitigate this event by placing control room HVAC in recirculation mode, and/or by using portable breathing equipment. In April 1992 DOE placed the operation of the facility on standby status pending definition of its long term mission. The NRC staff has conducted a safety review of the FFTF and concluded that it presents no undue risk to public health and safety beyond its site boundary consistent with safety consideration for light water reactors (NUREG-0358, supplement 1, May 1979). Thus, based on the above information, there are no nearby facilities within the 5 miles radius that could pose a potential threat to plant operations. Therefore, this event is screened from further evaluation.

5.4.7 Nearby Military Facilities

There are no military facilities within 5 miles of the WNP-2 site (FSAR 2.2.1). Therefore, any potential hazards associated with military facilities are screened from further evaluation.

5.4.8 Hazardous Material Releases From On-Site Storage

The WNP-2 design of habitability systems was reviewed by the NRC against the requirements of SRP Section 6.4 and found to comply with the SRP, and meet the intent of Regulatory Guide 1.78, GDC 4, and GDC 19. Therefore, this issue is screened from further consideration. In the following only a summary of considerations given for hazardous material release in the WNP-2 design is provided based on information contained in the FSAR and Reference 8.

As described in the "External Fires" section of this report, the control room habitability systems are designed to ensure habitability inside the main control room during all normal and abnormal events, including fires (internal and external), hazardous chemical releases, and 30 days of habitability following a LOCA (FSAR 6.4 and Reference 5.6.8). In the event of hazardous chemical release, the control room HVAC is designed to operate in recirculation mode without filtering by the control room emergency filter units. Permanent changes since the issuance of the OL to circulating water chemistry control system have eliminated the need for chlorine gas and the gas is no longer stored on the WNP-2 site. Also, chlorine is not used in other facilities (WNP-1 and DOE's FFTF) that are within five miles of WNP-2.

In accordance with Regulatory Guide 1.78, other on-site stored hazardous chemicals were reviewed to assess their potential impact on the WNP-2 control room in the event of a release. Cylinders containing nitrogen, a liquid storage tank, and cardox (CO^2) are stored near the control room. It was determined that in terms of possible asphyxiation, none of these chemicals are of sufficient quantity to pose any problem (FSAR 6.4.4.2.3). Also, potential effects of hydrogen gas stored in the gas bottle storage building and a trailer parked adjacent to the gas bottle storage building were found to be negligible. Based on the above information and the NRC acceptance of the design based on SRP requirements, events associated with hazardous material release from on-site storage can be screened from further evaluation.

5.5 Other External Events Considered for WNP-2

The design of the WNP-2 plant was reviewed by the NRC against the Code of Federal Regulations (CFR), the requirements of the construction permit and the NRC Standard Review Plan (SRP, NUREG-0800, July 1981 edition). The conclusions of the review are documented in the Safety Evaluation Report (SER) published by the NRC (NUREG-0892) in March 1982 and periodically amended thereafter. The SER was used as the primary document to assess the NRC's acceptance of various plant design features that are applicable to the other external events. In addition, other relevant licensing materials and site specific information were reviewed to identify any changes to design or assumptions as documented in the Final Safety Analysis Report (FSAR) since the operating license was issued. In the following, the progressive screening approach described in NUREG-1407 (Figure 5-1) is utilized to examine each of the possible events delineated in Section 2.0 as potential events (Reference 5.6.4).

5.5.1 Severe Temperature Transients
Extreme Heat and Cold

Potential Effects: Loss of ultimate heat sinks
Loss of offsite power (LOOP)

This portion of the evaluation concerns averages and extremes of climatic conditions which could affect the safety of the plant. Important equipment are, by design, protected from severe temperatures by placing them in appropriate structures, covered insulation, or heated/cooled areas to insure adequate heat transfer. Primarily, effects of these conditions are limited to the ultimate heat sink and to offsite power. The effect of weather on offsite power is included in the initiating frequency of a LOOP in the IPE study, based on six years of data following WNP-2's first year of commercial operation.

The 1981 SRP acceptance criteria (Section 2.3.1) is that regional meteorological conditions and phenomena meet the requirements of the following regulations:

- 10 CFR Part 50, Appendix A, General Design Criterion 2 (GDC 2), "Design Bases for Protection Against Natural Phenomena," with respect to information on severe regional weather phenomena that have historically been reported for the region and that are reflected in the design bases for structures, systems and components important to safety.
- 10 CFR Part 100, k100.10(c), with respect to the consideration that has been given to the regional meteorological characteristics of the site.

WNP-2 is located in a continental-type climate; as a result, the site experiences wide ranges and variations in annual temperature conditions. According to the data collected at the Hanford meteorological station, the temperature may range between -33°C (-27°F) to 46°C (115°F). During the design of WNP-2, detailed meteorological data for the site was used to evaluate the performance of the spray ponds with respect to (1) maximum evaporation and drift loss and (2) minimum water cooling. Using the meteorological data the ultimate heat sink was designed (Reference 5.6.5, Section 2.3.1.2.3, 9.2.5 and Appendix C) to satisfy the regulatory requirements of Ultimate Heat Sink (RG 1.27, Rev.1.). In a response to a NRC bulletin on "freeze protection of safety related process, instrumentation and sampling lines," the Supply System described the WNP-2 design to protect against cold temperature as follows:

All safety related process, instrument and sampling lines for WNP-2, with the exception of the standby service water system are contained entirely within heated buildings. Heating from existing heat sources will maintain building temperatures above freezing at the extreme cold weather condition of -27 degrees Fahrenheit.

Portions of the standby service water system piping, from the pumphouse to the reactor building and other heated areas of the plant, are routed outside. Most of this outside piping is buried well below the frost line. Sections of the piping not buried are either drained when water is not flowing through them or are insulated and heat traced (PPM 3.1.9).

In addition to these design features, WNP-2 has a procedure that deals with cold weather operations. The PPM provides specific information and guidance on implementation of actions to prevent damage from cold weather (e.g., draining of pipes/systems). This procedure is normally implemented by the scheduled maintenance system on November 1 and terminated on April 1. Detailed cold weather operation guidance is also provided within several plant procedures (PPMs) that deal with the Spray Ponds, the Cooling Tower, and the Emergency Diesel Generator systems. The current plant procedures and the sections that deal with cold weather conditions for these systems are PPM 2.4.5 Section 5.6, PPM 2.6.1 Section 5.5.3, and PPM 2.10.4 Section 5.3, respectively.

The NRC staff in its review of the WNP-2 design, as documented in the FSAR, concluded in the SER (Section 2.3.1) that, as per requirements of 10CFR100.10, adequate consideration has been given to the regional climatology. Furthermore, it concluded that the requirements in General Design Criteria (GDC) 2 have been met for meteorological parameters. Thus WNP-2 meets the applicable SRP criteria for these events and no further analysis of these events will be necessary.

5.5.2 Severe storms
Ice, Hail, Snow, Dust and Sand Storms

Potential Effects: LOOP
Degradation of Ventilation system and/or ultimate heat sink

Effects of these events are limited to impacts upon offsite power and to some extent to the ultimate heat sink and heating, ventilating and air conditioning (HVAC) systems. The regional meteorological conditions associated with these events were used as design and operating bases for WNP-2. The effect of weather on loss of offsite power is included in the initiating frequency of a LOOP calculated for the IPE, and is based on six years of data following WNP-2's first year of commercial operation. Safety related equipment and associated piping are protected from cold weather. Included in the design bases are dust storms which can occur at the site area during windy periods (FSAR and SER, Section 2.3.1). Based on measurements made at the Hanford Reservation a "worst case" dust storm with an average dust concentration of 8.9 mgm/m^3 with a duration of 18 hours was identified. However, the design basis volcanic ash fall has an average dust concentration of 174 mgm/m^3 and a 20-hour duration. Therefore, the volcanic ash fall is considered for worst case dust effects on safety related components such as the diesel exhaust. Detailed evaluations of potential effects of design basis ash fall on safety related systems and structures are discussed in Section 5.1 of this report.

The main control room, the cable spreading rooms, and the critical switchgear area are of primary concern if HVAC is sufficiently affected by severe storms. The control room at WNP-2 is equipped with two independent 100% capacity HVAC systems to handle air and particulate during emergencies. The control room habitability systems are designed to ensure habitability inside the main control room during all normal and abnormal events, including design basis ash fall. The effects of these events on the critical rooms/areas are negligible for the expected frequency and duration of these events as the rooms are enclosed and the total inside environment change of fresh air is within equipment capacity limits for maintaining inside design conditions (FSAR, Section 9.4).

Diesel generators are designed to withstand expected severe meteorological events for the WNP-2 site including rain, freezing rain, dust, snow, and cold weather operation with no load. In a letter to the NRC, the Supply System described this diesel generator capability to withstand such events so as to permit starting of the diesels.

The regulatory requirements (SRP) for severe storms are the same as those for the severe temperature transient described earlier. As concluded in the NRC's SER, WNP-2 meets the applicable SRP criteria for these events and no further analysis of these events will be necessary.

5.5.3 Lightning

Potential Effects: LOOP

Many complete and partial losses of offsite power events at nuclear power plants have involved lightning, but in most cases the power was restored within a short period of time. The events with the longest restoration times are for those that resulted in grid instability. WNP-2 is designed with protective relay schemes which function to mitigate damage during grid instabilities by automatically disconnecting electrical sources and loads until electrical grid stability is regained. For WNP-2, loss of offsite power means the power sources from the normal transformers, i.e., the startup transformer and the backup transformer, are lost. The effect of weather, including lightning, on loss of offsite power is included in the IPE initiating frequency for LOOP, and is based on six years of data following WNP-2's first year of commercial operation. In this six years of operation, there were no events involving a complete loss of offsite power during normal operation due to lightning or any other events. The probability of a lightning strike with significant impact on the site (no such event to date), coupled with the protective design features of the plant (including diesel generators), results in the conclusion that lightning is not a significant contributor to risk, and its impacts are already included within the IPE. Therefore, no further examination of this event is necessary.

5.5.4 External fires

Potential Effects: LOOP
Forced plant ventilation isolation
Evacuation of control room

External fires are fires occurring outside the plant site boundary, principally range fires. Other fires such as oil tank fires or fires due to nearby explosion are considered under the nearby military/ industrial accidents portion of the IPEEE. In the U.S., there has been at least one instance of loss of offsite power caused by an external fire: Pilgrim on May 1, 1977 experienced a partial loss of offsite power due to a nearby forest fire. At WNP-2, on August 12, 1984, an unusual event was declared due to a range fire that approached the plant site. The event was declared unusual due to the potential effects of the fire (which was approaching the plant boundary) on the offsite power supply. The fire was eventually extinguished by Department of Energy fire fighting crews. The unusual event was terminated at 8:00 p.m. At the time of this incident, WNP-2 was undergoing preoperational tests. If the fire was not extinguished, it is unlikely that it would have spread onsite due to clearing of the site during the construction stage. In areas adjacent to WNP-2, major buildings and auxiliary facilities are maintained to prevent weed growth by landscaping, ground cover, and weed control spraying (FSAR, Section 2.2.3.1).

The control room habitability systems are designed to ensure habitability inside the main control room during all normal and abnormal events, including fires (internal and external), hazardous chemical releases, and provide 30 days of habitability following a LOCA (FSAR, Section 6.2 and Reference 5.6.8). For external fires (and for hazardous chemical release), the control room HVAC system will be operated in the recirculation mode without filtering by the control room emergency filter units (CREFU). Smoke detectors are equipped in the control room fresh air intakes to alarm and enunciate in the control room. The smoke detectors are also electrically interlocked with the three-hour fire-rated dampers down stream of the detectors through an electrothermal link. The control room HVAC will be placed in a recirculation mode without filtration upon activation of the smoke detectors. Portable breathing apparatus is also provided in the control room for operating personnel protection (FSAR 6.2).

Based on the design features and preventive measures taken, external fires are considered not to pose a threat to safe operation of the WNP-2 plant. Therefore, no further examination of this event is required.

5.5.5 Extraterrestrial Activity

Potential Effects: Core Damage (Frequency 6E-8 to 7E-10 per Year, NUREG/CR-5042, Reference 12)

These events are considered to be natural satellites (such as meteors) or artificial satellites which enter the earth's atmosphere from space. In NUREG/CR-5042, it is estimated, based on analysis contained in Reference 5.6.13, that the probability of meteorites striking a nuclear power plant is between 7E-10 and 6E-8/reactor year. Thus, on the basis of the low probability, this event is screened out from further consideration.

5.5.6 Earth Movement (Avalanche, Landslide)

Potential Effects: Not Applicable (other than seismically induced, which is considered separately)

According to NUREG/CR-5042, avalanches are not a threat to any plant in the United States. Each nuclear plant site is extensively reviewed by the NRC at the site construction stage. Landslides and other large earth movements (other than those due to seismic events) are also not considered in detail for the WNP-2 site because it is located on level terrain and is not subject to these types of earth movements. Therefore, no further evaluation of this event is necessary.

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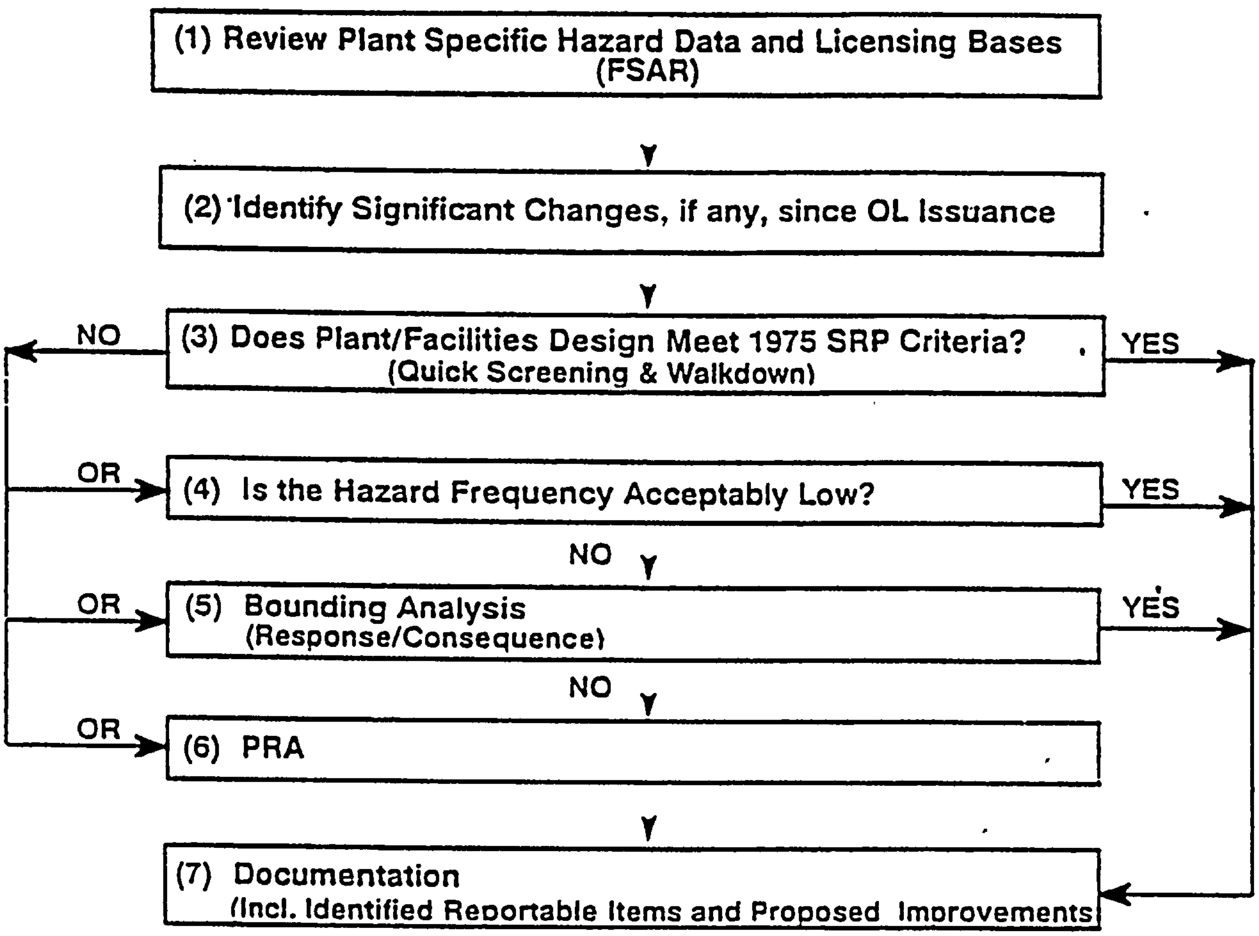

Figure 5-1 Recommended IPEEE Approach for Other External Events

DESIGN BASIS TORNADO CHARACTERISTICS

Region	Maximum Wind Speed [a] (mph)	Rotational Speed (mph)	Translational Speed (mph) Maximum	Translational Speed (mph) Minimum [b]	Radius of Maximum Rotational Speed (feet)	Pressure Drop (psi)	Rate of Pressure Drop (psi/sec)
I	360	290	70	5	150	3.0	2.0
II	300	240	60	5	150	2.25	1.2
III	240	190	50	5	150	1.5	0.6

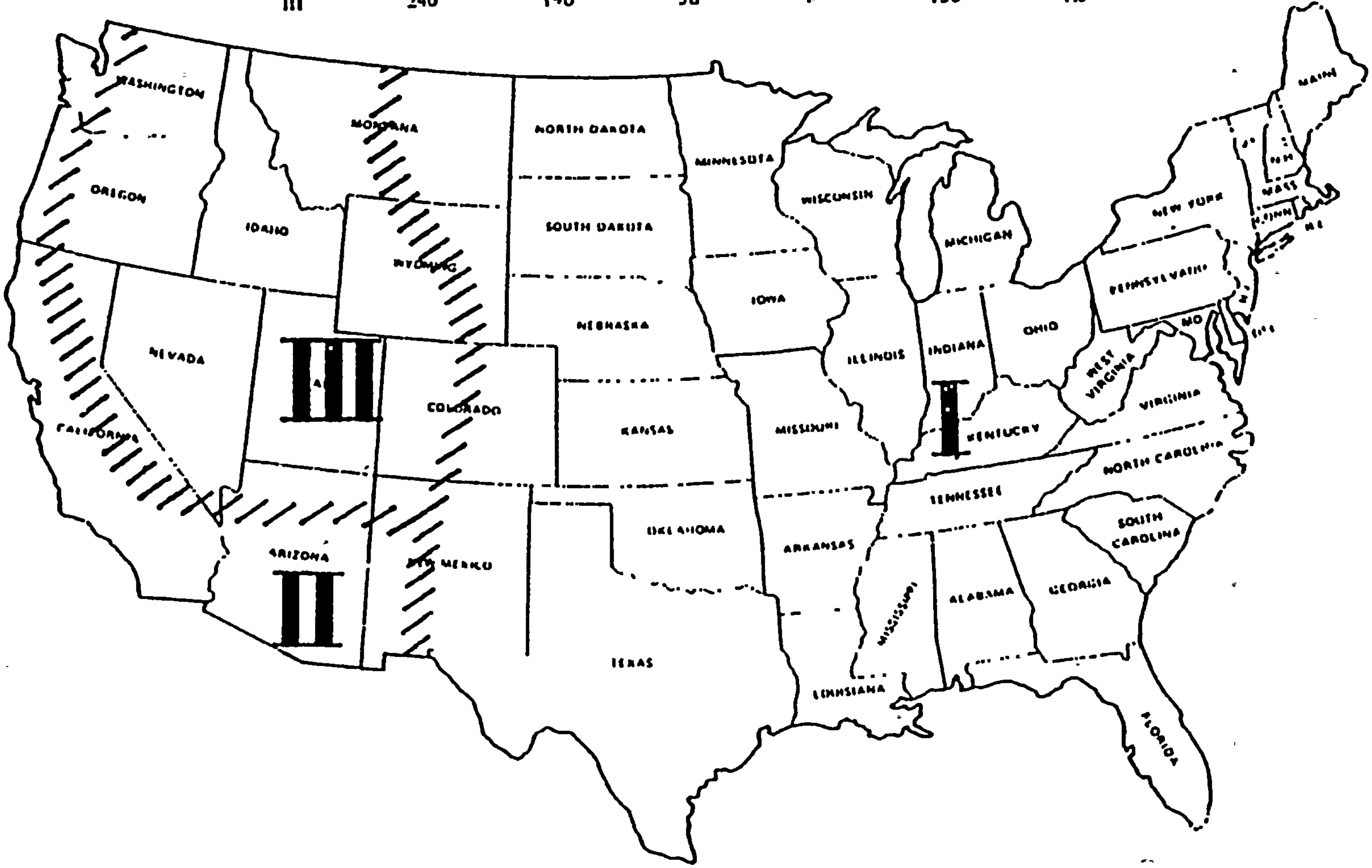

Figure 5-2 Tornado Intensity Regions

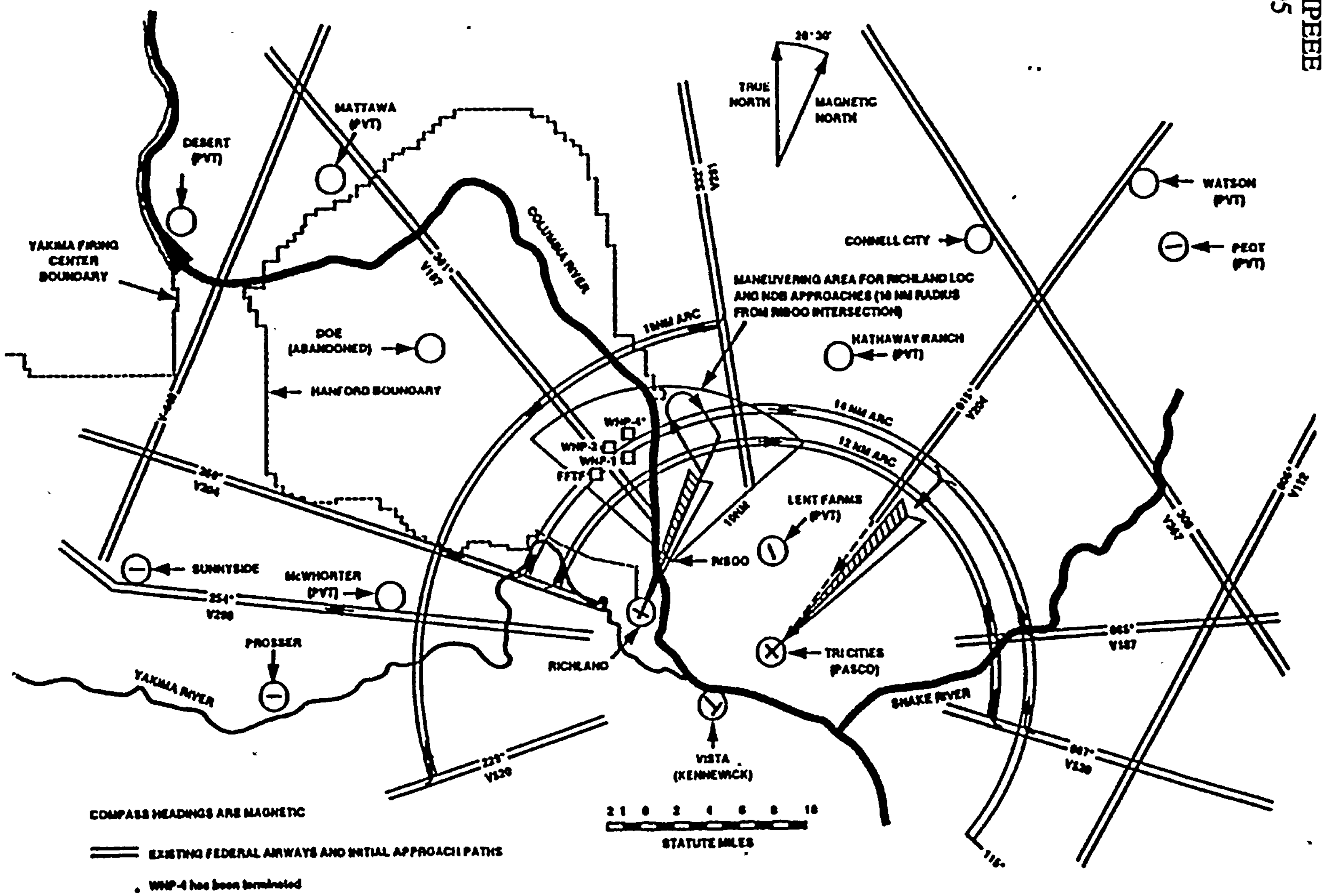
TRUE NORTH
MAGNETIC NORTH
20°30'
MATTAWA (PVT)
DESERT (PVT)
YAKIMA FIRING CENTER BOUNDARY
COLUMBIA RIVER
DOE (ABANDONED)
HANFORD BOUNDARY
CONNELL CITY
WATSON (PVT)
PEOT (PVT)
MANEUVERING AREA FOR RICHLAND LOC AND NDB APPROACHES (10 NM RADIUS FROM RISOO INTERSECTION)
HATHAWAY RANCH (PVT)
18 NM ARC
14 NM ARC
12 NM ARC
WNP-4*
WNP-3
WNP-1
FFTF
LENT FARMS (PVT)
RISOO
SUNNYSIDE
McWHORTER (PVT)
PROSSER
RICHLAND
TRI CITIES (PASCO)
VISTA (KENNEWICK)
YAKIMA RIVER
SHAKE RIVER
V187
V204
V298
V120
V112
COMPASS HEADINGS ARE MAGNETIC
EXISTING FEDERAL AIRWAYS AND INITIAL APPROACH PATHS
STATUTE MILES
* WNP-4 has been terminated

Figure 5-3

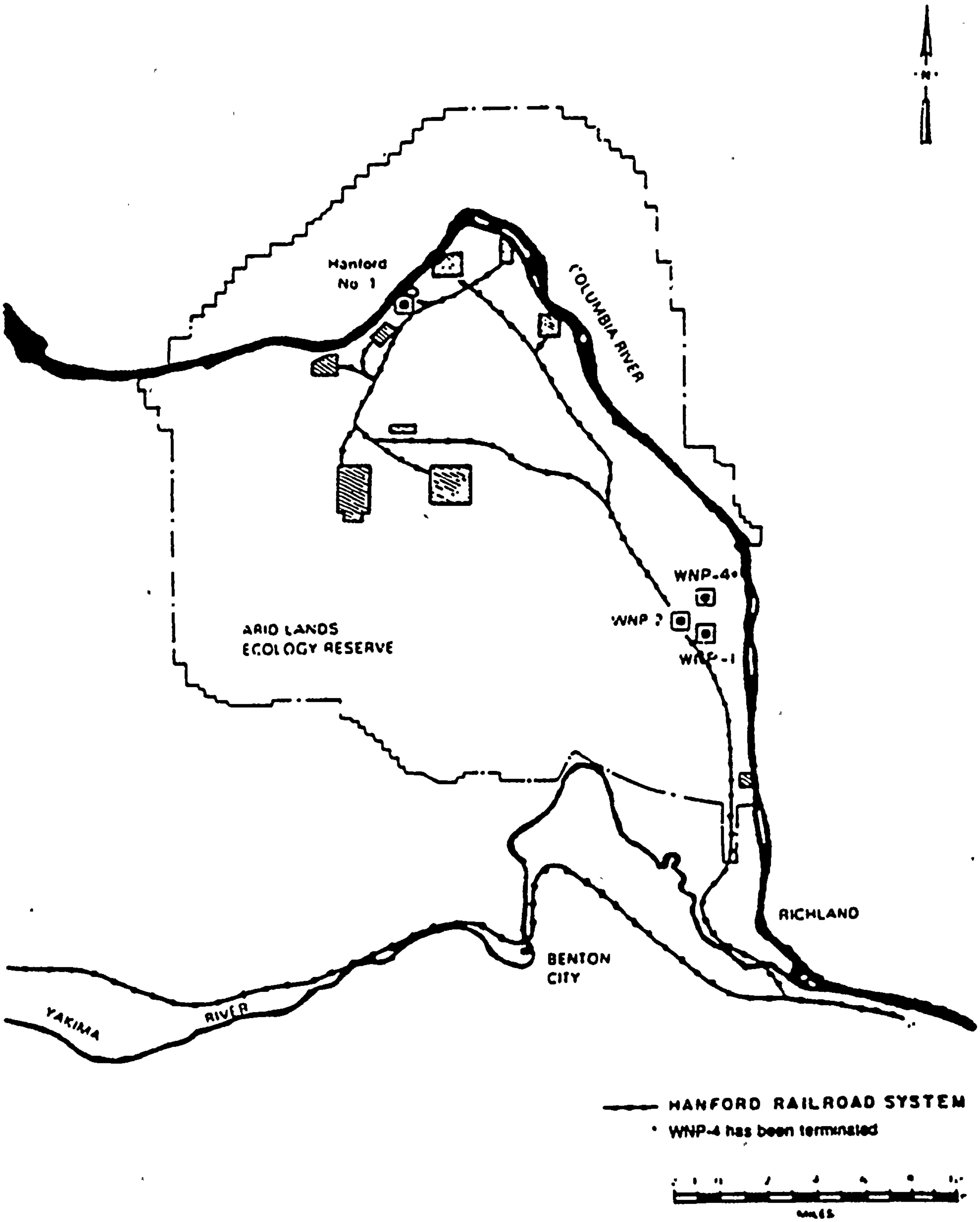

Figure 5-4 Hanford Reservation Railroad System

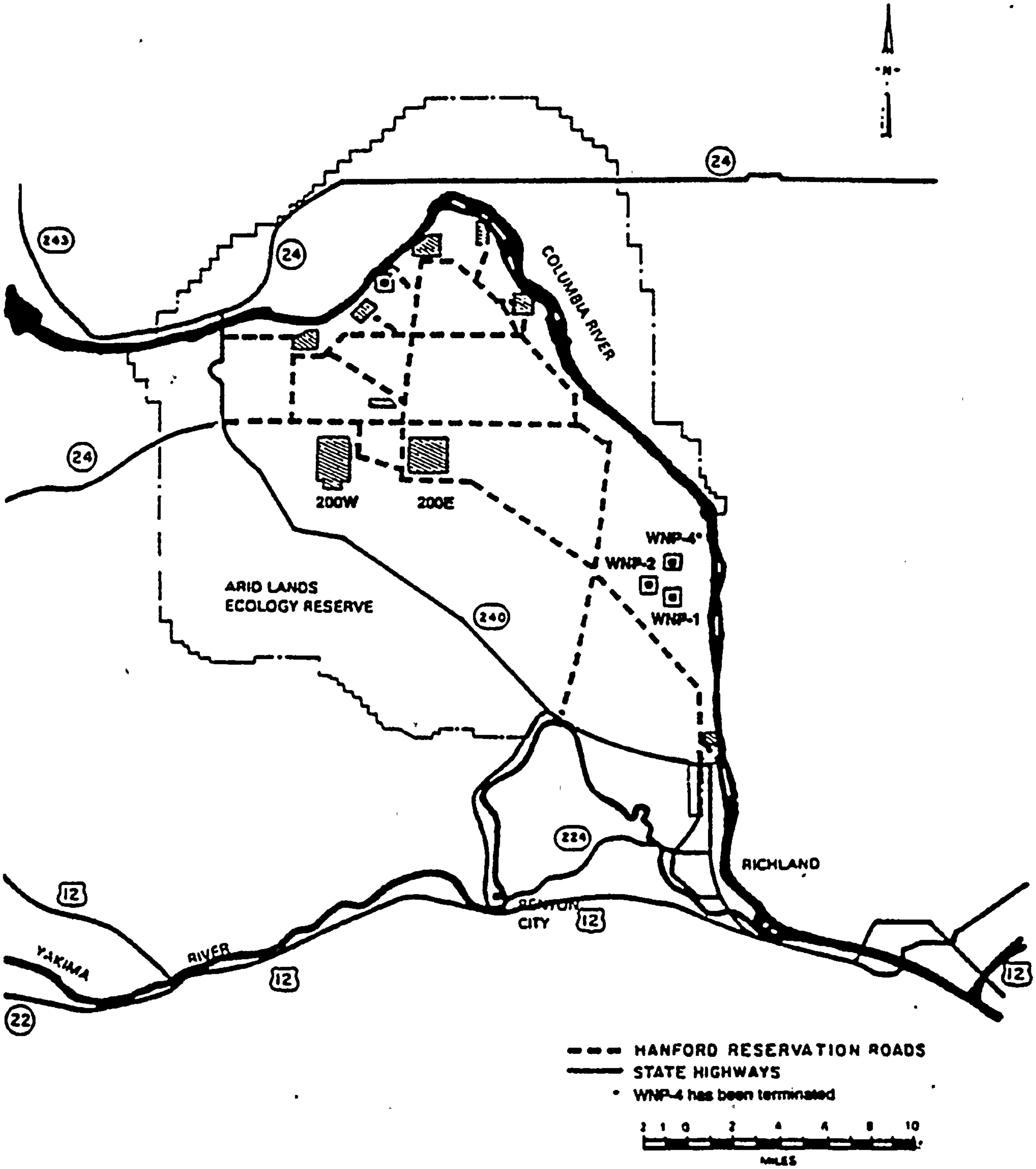

Figure 5-5 Hanford Reservation Road System

6.0 LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE Program Organization

In the early 1980s, the Supply System established a Senior Management Review Group to maintain cognizance of the severe accident issues, work with the industry's IDCOR Program through the policy group and with technical reviews, and assess potential impacts to WNP-2 as the NRC formulated the Severe Accident Policy and drafted the regulations that became Generic Letter 88-20 and supplements. The Supply System established a risk assessment team within the Engineering Directorate in anticipation of the release of Generic Letter 88-20. As the scope and cost of the individual plant examination using probabilistic risk assessment methodology became clear, the decision was made early to internalize the methodology to be able to benefit from the knowledge gained and apply it to other areas within the Supply System. Therefore, the same risk assessment team members involved in the original IPE and the revision to the IPE, are, for the most part, the team members involved in the IPEEE.

The risk assessment team members involved in the IPEEE are listed on the cover sheet of this report. In addition, a former Shift Manager from Operations was made available full-time during the initial information gathering tasks and for the seismic walkdowns. This early Operations support in the IPE and the IPEEE was a significant contributor to the quality of the analyses and results. From the list of contributors shown on the Acknowledgement page, it is obvious that all technical organizations of the Supply System contributed to this effort. The cost of the internal events IPE was approximately $3.2M and the cost of the external events IPEEE was approximately $3M including internal manpower, consulting services, and computer code acquisitions. The analysts assigned to this effort remained throughout the entire project providing the consistency and continuity necessary to bring the effort to fruition on schedule.

Major tasks performed by Supply System personnel included the system information gathering, system modelling and analysis, initiator data gathering, event tree preparation and quantification, and sensitivity analysis. Consulting firms, TENERA, L.P. (with principle subcontractor EQE Engineering) and VECTRA (formerly ABB Impell with principle subcontractor NUS Corp) were hired to assist in preparation of the IPEEE. The function of TENERA/EQE was to provide new site specific seismic hazards analysis, system and structure responses, provide SQUG qualified seismic walkdown team leaders, and provide initial component fragility curves. They also assisted with the seismic systems and sequence analysis tasks, and provided peer review of the IPEEE-Seismic products as they were generated. VECTRA/NUS Corp provided fire protection qualified engineers to perform system walkdowns, assist in cable routing studies, generate fire hazard initiation frequency analysis, and coordinate with TENERA/EQE on seismic-fire interaction issues. They also provided peer review of the IPEEE-Fire products as they were generated. TENERA, L.P. was contracted to review the WNP-2 FSAR and records to draft positions on the Other External Events.

6.2 Composition of Independent Review Team

The WNP-2 IPEEE has received a multi-tiered review in terms of technical review, peer review, independent in-house review, and management review. Technical reviews of the analysis were performed as the individual analyses were completed. If the contractor completed an analysis, it was reviewed by Supply System personnel prior to acceptance. If the Supply System completed an analysis, it was reviewed internally, as well as, by the appropriate contractor. The system analysis (fault tree) models, as modified for external events, were prepared in accordance with 10CFR50 Appendix B QA requirements and reviewed for reasonableness compared to other plants models and results. The COMPBRN IIIe model input files were also prepared to Appendix B requirements. Technical peer review of all quantification results and phenomenological results were performed by TENERA/EQE for seismic and VECTRA/NUS for fire.

An Independent Review Team was also established to review the WNP-2 IPEEE. This review team was composed of individuals not associated with the preparation of the IPEEE but who are experienced and knowledgeable of BWR systems, safety evaluations, design basis safety analyses, reliability methodology, IPE for External Events, and/or training in these functional areas. The Independent Review Team members and their organizational affiliation were:

R.J. Barbee, System Engineering; M. E. Kappl, Operations; J. T. Little, Maintenance; J. A. McDonald, Quality Assurance; L. D. Sharp, System Analysis; E. Tiedermann, Illinois Power; Craig Nierode, NSP.

The Independent Review Team met once. This session was conducted to a) respond to concerns/questions raised during the members initial review, b) have a member of the IPEEE team introduce topics of interest, e.g., seismic or fire fault tree modelling, c) present the results in the appropriate section of the IPEEE Report, and d) respond to questions or record comments for resolution. This review process was used for all aspects of this report.

In addition to the Independent Review Team, TENERA and NUS reviewed a draft of the original report and provided comments. Their review was comprehensive in terms of ensuring the individual tasks and results were properly collated into this report and by performing a peer review of other plant's IPEEE Reports. Presentations have been made to the Corporate Nuclear Safety Review Board and to the Supply System's Senior Managers to discuss the methodology and results of the IPEEE analysis. These presentations focused on what parameters most affect the CDF, what can be done about them, and the tasks remaining to complete the Severe Accident Program. The potential uses of the methodology and IPEEE results in other Supply System processes were also discussed.

6.3 Areas of Review and Major Comments

The WNP-2 IPEEE Report, Revision 0, was reviewed by the Independent Review Team and by Tenera, L.P. These reviews were conducted at various stages of report completeness to ensure resolution of comments addressing technical correctness and issues to report content and clarity. The number and content of the comments are too numerous to include in this report. In all cases, however, satisfactory resolution was achieved between the PRA team and the reviewers.

7.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

The IPEEE study results show that internal fire is the largest contributor to plant risk. The fire areas which produce the most important sequences are the electrical equipment rooms, RC-4 and RC-7, the 471' elevation of the reactor building, and the turbine generator corridor. As with the IPE study performed earlier, one of the major beneficial impacts of performing the IPEEE has been the gain in knowledge of major features and insights to plant system response to externally initiated accidents. Therefore, a general recommendation is to disseminate this knowledge to the organizations of the Supply System that can utilize the knowledge in their day-to-day activity.

7.1 Unique Safety Features of WNP-2

General Seismic Ruggedness - The design of WNP-2 was found to be very conservative with respect to earthquake. Even using an RLE of 0.5g, which is twice the DBE peak ground acceleration, very few items important to safety did not screen away. All building structures were found to have a large measure of conservatism in design and were screened out in the RLE.

Site Location - The WNP-2 site location eliminates or reduces the severity of many external events. The distance from the nearest natural body of water eliminates the potential threat of many external flooding sources. Even catastrophic dam break studies show conservative peak water levels to be well below plant grade. Risk due to high wind, air travel, lightning strike, nearby facilities, severe storms, and precipitation extremes are all relatively low for a U.S. plant site.

7.2 Walkdown Findings

Several items were noted during plant walkdowns that presented potential vulnerabilities. These items were brought to the attention of the management as they were discovered and corrected via normal plant procedures. None of the items found during the walkdowns were found to result in an plant condition outside the design envelope.

Two air handling units in the Division 1 diesel generator room were noted to be missing anchorage nuts or washers. Subsequent analysis showed that the units were capable of withstanding DBE loads and perform their intended functions in the as found state. Both air handling units were restored to design anchorage configuration. The Division 2 units were thoroughly inspected and found to be installed as designed.

During the seismic walkdown on E-SM-7 and E-SM-7/75/2, the connection between these cabinets was found to be less than expected. Rather than being connected at the edges, as is usually the case, these cabinets were connected at the center of the adjoining panel. This left the cabinets susceptible to banging during a seismic event. Subsequent analysis showed that banging induced relay chatter did not impact safety functions during a seismic event.

Nonetheless, these cabinets will be edge connected the next time such activities can be safely performed.

Three motor control centers (MCC) and two instrument racks important to safety were found to have hangers installed in close enough proximity to question the affect that seismic induced banging might have on operability during an earthquake. All five situations were found to be operable for all safety related functions. Nonetheless, each of these potential banging situations will be remedied via normal plant processes.

7.3 IPEEE Recommended Improvements

7.3.1 Hardware Improvements

FP Diesel Engine Batteries - During the walkdown for seismic-fire interactions, it was noted that while the diesel powered fire pump installed at WNP-2 was very ruggedly mounted, the batteries were not tied down. Thus, while the pump, controls, engine, and fuel storage tank would remain intact for any probable seismic event, the diesel engine may not start because the batteries could topple. This pump is not required to withstand earthquake events per the WNP-2 license. However, earthquakes often result in fire damage. No situations were noted within the plant for seismic induced fire concerns, but the outlying buildings may be at risk. Actions are being taken as good business practice to tie down the batteries for the diesel powered fire pump. This is an inexpensive upgrade that will increase the availability of the fire water system in an earthquake situation.

MCC Base Connections - The major contributing sequences to seismic risk at WNP-2 involve failure of MCC base connections. These cabinets currently meet design basis requirements and do not pose an undue safety hazard. However, it is recommended that the costs and benefits of increasing the seismic capability of these cabinets be explored. This study should be coordinated with the study of the switchgear room cooling study discussed in section 7.3.2.

7.3.2 Procedure Based Improvements

Transient Combustible Limits - In general, combustible loadings in critical plant areas are low and are installed or constructed in a manner that make fire propagation improbable. However, several areas (most notably the cable spreading room and the cable chase area) are very sensitive to transient combustible loadings. Fire simulation studies show that large quantities of transient combustibles, strategically located in these areas can cause multi-division damage and have a large impact on plant risk. It is recommended that existing procedures for control of transient combustibles in these areas be reviewed for adequacy. If determined to be necessary additional transient combustible procedures should be developed and implemented for these areas.

Fire Suppression Loop Isolation Valves - The fire suppression system at WNP-2 provides water via a large main piping loop to both the plant and its outlying buildings. The outlying buildings at WNP-2 are not designed to the same seismic standards as the plant and are considerably more susceptible to damage in an earthquake. Multiple fire system breaks in outlying buildings may limit the availability of water to the plant proper.

As stated earlier, no specific instances of seismic induced fire vulnerability were noted during walkdowns at WNP-2. No hardware changes are recommended to deal with this situation. However, information concerning the location of shutoff valves for fire suppression system connections between outlying buildings and the main fire suppression loop may be important following a seismic event. Existing procedures and training were evaluated and a determination was made that such information is available to operations in a format that is easy to decipher both in terms of valve location and building(s) served.

Alternate Switchgear Room Cooling - One of the most important functions impacted by MCC failure in very large seismic events is room cooling to the critical switchgear rooms. It is recommended that the costs and benefits of providing procedural direction for opening the doors to a critical switchgear room be explored as an alternate means of providing adequate cooling to the area. This activity should be coordinated with the MCC base connection study in section 7.3.1 because increasing the seismic capacity of E-MC-7F and 8F will dramatically reduce the need for this study.

Inadvertent HALON System Actuation - During the relay chatter evaluation, nine relays of unknown seismic ruggedness were revealed in the fire protection system. Should these relays chatter during a severe seismic event, the Halon system in the control room could inadvertently actuate. This could lead the control room personnel to interpret that a fire is present. The Training Department has been informed of this potential problem. Control room crews have been advised to take further steps to confirm an actual fire exists before accepting the HALON actuation at face value.

Critical AC Bus Alternate Power - The recovery of the critical AC Buses SM-7 and SM-8 was shown to be significant in reducing fire induced CDF. This recovery action is proceduralized and due to its importance it is recommended that specific training scenarios be included in the operator training program.

8.0 SUMMARY AND CONCLUSIONS

The Supply System has performed an Individual Plant Examination for External Events of WNP-2 using latest plant system and plant site information, procedures, accident initiator data, and component failure data. The general methodology used in this report follows the guidelines described in NUREG 1407, "Procedural and Submittal Guidance for the IPEEE for Severe Accident Vulnerabilities." PSA analyses were performed for internal fire and seismic events. The progressive screening approach recommended in GL 88-20 was used to assess the remaining events. Plant specific data were used whenever possible. However, for fire ignition frequency calculations, generic U.S. nuclear industry data was used in the form of the fire events data base, NSAC 178L. Core damage frequency was quantified using the NUPRA computer code for the fire PSA and a combination of NUPRA and EQESRA computer codes were used for the seismic PSA. The system interactions and dependencies were accounted for within the analyses which were based on the model constructed for the IPE submittal.

Supply System staff were involved in all aspects of the IPEEE. This includes information gathering, walkdowns, fire propagation analysis, fault tree modification, event tree preparation and quantification. The seismic hazard curve was developed by GEOMATRIX and the plant equipment fragility curves were developed by EQE International. The Supply System performed a review function only for these aspects. In addition, VECTRA and NUS assisted with all portions of the fire PSA, and TENERA assisted with all aspects of the seismic PSA and the other events evaluation. The IPEEE has received multi-tiered review in terms of technical review, peer review, independent in-house review, and management review.

The core damage frequency for fire outside the Control Room was calculated to be a mean value of 9.16E-6/year. The sequences are dominated by fires in areas RC-4, RC-7, R-Id, TG-Ia, and TG-Ik. The Control Room fire adds a CDF of 8.4E-6/year.

The seismic contribution to core damage frequency was calculated to be a mean value of 2.1E-5/year. The seismic sequences are dominated by seismic induced loss of offsite power sequences.

Using the progressive screening approach high winds, tornadoes, external floods, transportation accidents, nearby facility accidents, severe storms, landslides, lightning strikes, and volcanoes were all shown to contribute less than 1.0E-6 to core damage frequency.