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ANALYSIS OF CORE DAMAGE FREQUENCY
DUE TO EXTERNAL EVENTS
AT THE DOE N-REACTOR

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ABSTRACT

A complete external events probabilistic risk assessment has been performed for the N-Reactor power plant, making full use of all insights gained during the past ten years' developments in risk assessment methodologies. A detailed screening analysis was performed which showed that all external events had negligible contribution to core damage frequency except fires, seismic events, and external flooding. A limited scope analysis of the external flooding risk indicated that it is not a major risk contributor. Detailed analyses of the fire and seismic risks resulted in total (mean) core damage frequencies of $1.96\text{E-}5$ and $4.60\text{E-}05$ per reactor year, respectively.

Detailed uncertainty analyses were performed for both fire and seismic risks. These results show that the core damage frequency profile for these events is comparable to that found for existing commercial power plants if proposed fixes are completed as part of the restart program.

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EXECUTIVE SUMMARY

In May 1986, a Level 1 Probabilistic Risk Assessment (PRA) was initiated on the N-Reactor, the only large water-cooled, graphite-moderated reactor in operation in the United States. After the Chernobyl incident, the scope of the study was expanded to include Level 2/3 studies with treatment of external events.

A Level 1 PRA consists of an analysis of plant design and operation focusing on accident sequences that could lead to core damage, their basic causes, and frequencies. Two kinds of accident initiators are considered for a Level 1 PRA, initiating events that occur within the power plant systems themselves and accident initiators caused by events external to the power plant systems. Examples of external initiators include earthquakes, floods and high winds. For any given plant, the expected core damage frequency is the sum of the frequencies due to internal and external events. The results of both analyses provide assessments of plant safety, design and procedural adequacy, and insight into how the plant functions from the perspective of preventing core damage. This report documents work performed for the Level 1 external events PRA. It describes the methodology used, assumptions, data and models that provide the basis for the work, and the final results.

The N-Reactor external events PRA fulfilled four primary objectives:

- a. Evaluation of the contribution of external events to plant core damage frequency
- b. Identification of potential plant modifications to reduce expected risk of operation
- c. Evaluation of dominant external event-initiated N-Reactor accident sequences and frequencies for input into the Level 2 analyses
- d. Technology transfer of the external events PRA methodology to Westinghouse Hanford Company (WHC)

Work was initiated with a systematic screening of the full range of external events to which the plant would be exposed. It was assumed that detailed seismic and fire risk analyses would be conducted. In addition, the following events were found to be potential contributors to the plant risk; extreme winds and tornadoes, external flooding, and internal flooding.

Bounding analyses were conducted for each of these events. External winds and tornadoes and internal flooding were eliminated from further consideration. However, external flooding due to random or seismically induced dam failure may be a significant contributor to core damage frequency. Results of the bounding analyses, systems analysis, and discussions of known conservatisms for external flooding are presented in Chapter 3.

Detailed system models were developed for those hazards which survived initial screening. These models were based on the final internal events system analysis (event trees and fault trees). Computer-aided screening techniques and generic failure data were used prior to detailed component failure analyses to eliminate unimportant systems and components and hence, reduce the calculational effort of the study. The data base used for the final quantification included plant specific data from the Level 1 PRA, additional site specific data such as the local seismic hazard curve and building fragilities, and generic data.

Several levels of review were implemented during the course of the analyses. These included:

- a. Independent review of the methodology and results by Sandia National Laboratory as part of their peer review process
- b. Review of the external events methodology and results by the WHC Level 1 PRA team
- c. Peer review of the program methodology and results by a panel of external PRA experts chaired by P. R. Davis of PRD Consulting
- d. Independent review of the methodology and results by WHC nuclear safety

Taken together, these review efforts provided a high level of confidence in the conclusions and results of this study.

One primary goal of an external events PRA is estimation of the frequency of all events resulting in core damage. The detailed screening showed that all external events had negligible contributions to core damage frequency except internal fire, seismic events, and external flooding. Analysis of the fire risk resulted in a total (mean) core damage frequency of $1.96\text{E-}5$ per year. Similar detailed analysis of seismic risk resulted in a total (mean) core damage frequency of $4.60\text{E-}5$ per year.

External events core damage frequencies are compared to the N-Reactor internal events core damage frequency in Figure 1. The upper and lower boundaries of each bar represent the 5th and 95th percentiles of the distributions. Thus, the bars include the central 90 percent of the distributions (note that the distributions are not uniform within the bars). The means and median of each distribution are also shown. Detailed uncertainty analyses were performed for internal events, fire and seismic core damage estimates. During the course of the seismic and fire risk analyses several important plant-specific vulnerabilities were discovered. Engineering changes to eliminate these vulnerabilities have already been incorporated into future operations plans for N-Reactor. The fire analysis base case assumes that the engineering changes described in Chapter 5 will be implemented prior to plant restart. As a sensitivity study the current (as is) plant configuration fire risk is also given. Detailed discussion of the effect of seismic modifications is given in Chapter 4. The risk decrease due to these seismic modifications is shown as on the dotted box in Figure 1.

Figure 1. N-Reactor core damage frequency ranges (5th and 95th percentiles)

1.0 INTRODUCTION

1.1 The N-Reactor Risk Analyses

This report describes the external events risk analyses for the N-reactor nuclear plant. The procedures used here have been applied (in whole or in part) to six power plants as part of the U.S. NRC-sponsored Unresolved Safety Issue A-45 resolution program (Ref. 1), to the Peach Bottom and Surry reactor plants as part of NUREG-1150, and to the Savannah River Department of Energy reactor facility. The simplified methodology utilized here is suitable for a wide range of nuclear safety applications. A full description of these procedures is given in Bohn and Lambright (Ref. 2). The simplified external event risk assessments described here provide for minimization of execution time and cost reductions through the use of past PRA experience, generic data bases and defensible methodological simplification. In addition, they meet the following objectives:

- a. To be consistent with internal event analyses. The same event trees/fault trees and random, common mode failure and test maintenance data will be used
- b. To be transparent. A standard report format should enable the reader to reproduce any of the results
- c. To be realistic. Best estimate data and models will be used as much as possible. Important plant-specific failure modes will be analyzed
- d. To be comparable. The external event analyses will be directly comparable with internal event analyses due to common generic data, common methodology, and common level of detail and presentation of results.

1.2 The Simplified External Event Methodology

The simplified PRA procedures described in this section are based on the following general concepts:

- a. The external event analyses are based on the internal event risk assessment plant system models and fault trees, and (other than preliminary data gathering) are not started until the internal events systems analysis (event trees and fault trees) has been finalized.
- b. Vigorous and systematic screening of the full range of external events to which the plant could conceivably be exposed (e.g., aircraft crash, external flooding, tornado, extreme wind, etc.) is performed to eliminate early all unimportant contributing events.
- c. Simultaneous and coordinated evaluation of all non-negligible external events is performed to minimize data-gathering efforts and prevent duplication of effort. For example, building fragilities for extreme winds can be derived directly from seismic fragilities. Also, simultaneous evaluation produces insights into interactions (for example, seismic-fire interactions) not otherwise readily perceived.

- d. In the analysis of each type of external event, computer-aided screening techniques and generic failure data are used prior to detailed component failure analysis calculations.

The general steps in the analysis of any external event hazard are shown below:

- a. Determine the hazard
- b. Model plant and systems
- c. Solve fault trees with screening techniques to determine non-negligible cut sets
- d. Determine responses, fragilities, and correlation for basic events in non-negligible cut sets
- e. Evaluate point estimate sequence and core damage frequencies
- f. Perform uncertainty analysis and sensitivity studies

These general steps apply to the full range of external events to which a power plant may be exposed. Table 1.1 presents a reasonably complete list of such events. Past PRA experience (Refs. 3,4) has shown that only a very few of these are significant contributors to risk at any given site. In fact, the seismic and fire events are commonly the most important contributors. In addition, external flooding, tornado or aircraft crashes are contributors though less frequent (and usually less significant).

Table 1.1

List of External Events

<u>Major PRA Consideration</u>	<u>Minor PRA Consideration</u>
Seismic	Lightning
Fire	Low Lake/River Level
Internal Flood	Ice Cover
	Avalanche
<u>Occasional PRA Consideration</u>	Forest Fire
External Flood	Industrial Facility Accident
Transportation Accidents	Landslide
Pipe Line Accidents	Meteorite
Aircraft Impact	Volcanic Activity
Extreme Winds	Hail
Tornado	

Simplifications in Step a, hazard determination, have been identified for both the seismic and fire analyses. Computer-aided screening techniques are used in Step c for fire, flood and seismic analyses to reduce the required number of plant-specific component failure calculations. For Step d, response determination, seismic design fixed-base structural models are utilized in conjunction with an accurate soil-structure interaction model. While not a simplification, this process has been made very efficient by standardization, and use of variabilities and correlation factors derived from previous detailed seismic PRA work. Thus, simplifications have been identified which result in a cost effective yet defensible analysis.

1.3 Steps in The Analyses

1.3.1 Plant Walkdown and Data Gathering

The N-reactor external events analysis began with a plant visit in March 1988. The initial visit served as the basis for the initial plant information request submittal. Prior to the first plant visit, the external events team was briefed by an internal events systems analyst as to the general character of safety systems, support systems, system success criteria and critical interdependencies. In addition, applicable N-Reactor Updated Safety Analysis Report (Ref. 5) sections were reviewed, and a basic set of plant general arrangement drawings were obtained for each team member.

The team consisted of the following personnel:

- PRA Project Manager - M. P. Bohn
- Team Leader - J. A. Lambright
- Seismic Structural Fragility Analysts - J. J. Johnson,
P. O. Hashimoto
- Fire PRA Analyst - J. A. Lambright
- "Other" External Event Screening Analyst* - R. Ravindra

During the initial walkdown, team members visited all areas containing safety or support equipment. Two full days were adequate for this initial visit. At the completion of this initial visit, the following had been obtained:

*Responsible for performing screening analysis for all external events other than seismic and fire.

- a. A list of components suspected of being vulnerable to seismic damage and requiring site-specific fragility analysis
- b. A list of potential secondary seismic structural failures (masonry walls, etc.) and components potentially damaged by these secondary failures
- c. A copy of the civil/structural drawing index for the plant from which needed drawings could be identified
- d. Sketches of typical anchorage details for important tanks, heat exchangers, electrical cabinets, etc.
- e. A visual evaluation of structural connectivity of floor slabs, wall-to-ceiling connections, location of diaphragm cut-outs etc., which define load-carrying paths. These were to be compared with structural drawings later.
- f. For each room or compartment containing essential safety equipment, an identification of fire sources (power cables, pump motors, solvents and other transient combustibles, etc.), locations of fire barriers, fire/smoke detectors, separation of cable trains etc. and a list of equipment in the room
- g. For each room or compartment, an identification of flooding sources (tanks, high or low pressure piping), floor drains, pumps, flood walls, flood detectors, etc.
- h. A brief list of key plant personnel or utility engineering/licensing personnel to be contacted later if specific questions arose

Following the initial plant visit, a list of needed drawings and documentation was prepared and sent to the designated plant contact. A second visit to the plant was made by the fire analysis personnel to allow for cable path tracing and verification. A final plant visit was made in March 1989. During this final visit initial conclusions as to plant vulnerabilities were reviewed with plant personnel, assumptions were verified, and final required data was obtained.

1.3.2 Screening of Other External Events

As mentioned in Section 1.1, the full range of possible external events was considered, but based on Reference 5 and the initial plant visit, the vast majority of the external hazards was shown to have negligible impact. The set of general screening criteria which was used is

documented in the PRA Procedures Guide (Ref. 6) and is summarized as follows:

An external event is excluded if:

- a. The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular external event. For example, it is shown by Kennedy, Blejwas and Bennett (Ref. 7) that safety-related structures designed for earthquake and tornado loadings can safely withstand a 3.0 psi static pressure from explosions. Hence, if the PRA analyst demonstrates that the overpressure resulting from explosions at a source (e.g., railroad, highway or industrial facility) cannot exceed 3 psi, these postulated explosions need not be considered.
- b. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is less than some small fraction of those for other events. In this case, the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as not significantly influencing the total risk.
- c. The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions and earthquake fault ruptures.
- d. The event is included in the definition of another event. For example, storm surges and seiches are included in external flooding. Another example is the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

These criteria are usually sufficient to exclude all but a few "other" external events. For those remaining, a simple bounding analysis (Ref. 8) will often provide sufficient justification for exclusion. The screening and bounding analyses for N-Reactor are given in Chapter 3.

1.3.3 Seismic Risk Assessment Methodology

A nuclear power plant is designed to ensure the survival of all buildings and emergency safety equipment in a worst-case ("safe shutdown") earthquake. The assumptions underlying this design process are deterministic and subject to considerable uncertainty. It is not possible, for example, to accurately predict the worst earthquake that will occur at the given site. Soil properties, mechanical properties of buildings, and damping in buildings and internal structures also vary significantly. To model and analyze the coupled phenomena that contribute to the total risk of radioactive release requires

consideration of all significant sources of uncertainty as well as all significant interactions. Total risk is then obtained by considering the entire spectrum of possible earthquakes and integrating their calculated consequences. This point underscores an important requirement for a seismic PRA; the nuclear power plant must be examined in its entirety, as a system.

A second important aspect which must be addressed in a seismic PRA is that, during an earthquake, all parts of the plant are excited simultaneously. Thus, redundant safety system components experience highly correlated base motion, and there is a high likelihood that multiple redundant components would be damaged if one is. Hence, the planned-for redundancy would be compromised during earthquakes. During earthquakes this "common-cause" failure possibility represents a potentially significant risk to nuclear power plants.

The simplified seismic risk methodology reported here is based, in part, on the results of two earlier Nuclear Regulatory Commission (NRC) sponsored programs. The first was the Seismic Safety Margins Research Program. In the SSMRP, a detailed seismic risk assessment methodology was developed. This program culminated in a detailed evaluation of the seismic risk at the Zion nuclear power station (Ref. 9). In this evaluation, an attempt was made to accurately compute the responses of walls and floor slabs in the Zion structures, moments in the important piping systems, accelerations of all important valves, and the spectral accelerations at each safety system component (pump, electrical bus, motor control center, etc.). Correlation between the responses of all components was computed from the detailed dynamic response calculations. The important safety and auxiliary systems functions were analyzed, and fault trees were developed which traced failure down to the individual component level. Event trees related the system failures to accident sequences and radioactive release modes. Using these detailed models and calculations, it was possible to evaluate the seismic risk at Zion, and determine quantitatively the risk importance of the components, initiating events, and accident sequences.

The second program was the NRC-sponsored Eastern Seismic Hazard Characterization program (Ref. 10) which involved a detailed earthquake hazard assessment of ten sites east of the Rocky Mountains. In the application to N-Reactor, this latter program was only used to determine uncertainty bands for the hazard curves. Results of these two programs formed the basis for a number of simplifications used in the seismic methodology reported here.

There are seven steps required for calculating the seismic risk at a nuclear power plant:

- a. Determine the local earthquake hazard (hazard curve and site spectra)
- b. Identify accident scenarios for the plant which lead to radioactive release (initiating events and event trees)

- c. Determine failure modes for the plant safety and support systems (fault trees)
- d. Determine the responses (accelerations or forces) of all structures and components (for each earthquake level)
- e. Determine fragilities (probabilistic failure criteria) for the important structures and components
- f. Compute the probability of core damage using the information from Steps a through e
- g. Estimate uncertainty in the core damage frequencies

Only the level of detail differentiates a simplified seismic analysis from a detailed seismic PRA. The seven steps of the simplified seismic risk analysis procedure are summarized below.

Step a - Seismic Hazard Characterization

Characterization of the seismic hazard at the N-Reactor Facility is based on existing hazard studies which have been performed both for N-Reactor and other nearby facilities. In particular, the recent study by Woodward-Clyde Inc. (Ref. 11) has summarized and updated the previous seismic studies applicable to the site.

Step b - Initiating Events and Event Trees

The scope of the N-Reactor seismic analysis includes all potential initiating events, including loss-of-coolant accidents (process tube rupture, large LOCA, small LOCA) and transient events. The event trees derived for the internal event analyses are utilized.

Step c - Fault Trees

Fault trees for the safety systems at N-Reactor have been developed for random failures only. These fault trees are used, with some modification to include basic events for seismic failure modes and to resolve the trees for pertinent cut sets to be included in the probabilistic calculations. Probabilistic culling is used in resolving these trees in such a way as to assure that important correlated failure modes are not lost.

Step d - Component and Structure Failure Descriptions

Component seismic fragilities are obtained both from a generic fragility data base and from plant-specific fragilities developed for components identified during the plant walk-down.

A generic data base of fragility functions for seismically-induced failures was originally developed as part of the SSMRP (Ref. 9). Fragility functions for the generic categories were developed based on a

combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Subsystem Hardness Assurance Program (Ref. 12). These data were statistically combined with an expert opinion data survey to produce fragility curves for each of the generic component categories reported in Reference 9. This generic data base was then updated by an evaluation of 19 site-specific seismic PRAs to yield the final generic fragility data base used for the N-Reactor seismic PRA.

Detailed structural fragility analyses were performed for all important safety-related structures at N-Reactor. These were included directly in the risk assessment.

Step e - Seismic Response of Structures and Components

Building and component seismic responses are estimated from peak ground accelerations at several probability intervals on the hazard curve. Three basic aspects of seismic response--best estimates, variability, and correlation--are generated. The Zion nuclear power plant analysis results from the SSMRP and simplified methods studies form the basis for assigning scaling, variability and correlation of responses.

In each case, SHAKE code (Ref. 13) calculations are performed to assess the effect of the local soil column (if any) on the surface peak ground acceleration and soil structure interactions. This permits an evaluation of the effects of non-homogeneous underlying soil conditions which can strongly affect the building responses.

Fixed base mass-spring (eigen-system) models are either obtained from the plant's architect/engineer or are developed from plant drawings as needed. Using these models one can compute the floor slab accelerations using the CLASSI code (Ref. 14). This code takes a fixed-base eigen-system model of the structure and input-specified frequency dependent soil impedances and computes the structural responses (as well as variation in structural response if desired).

Variability in responses (floor and spectral accelerations) is assigned based on the SSMRP results. The recommended uncertainties (expressed as standard deviations of the logarithms of the responses) are shown below:

<u>Quantity</u>	<u>Random</u>
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

Correlation between component failures has been included explicitly. In computing the correlation between component failures (in order to quantify the cut sets) it is necessary to consider correlations both in

the responses and in the fragilities of each component. Inasmuch as there are no data as yet on correlation between fragilities, the fragility correlations between like components are taken as zero, and the possible effect of such correlation is quantified in a sensitivity study. The correlation between responses is assigned according to a set of rules that are explained in Chapter 4.

Step f - Probabilistic Failure and Core Damage Calculations

Given the input from the five steps above, the SETS code (Ref. 15) is used to calculate the required output (probabilities of failure, core damage, etc.).

Step g - Estimate Uncertainties

Both point estimates (mean and median) as well as complete uncertainty distributions were computed for all accident sequences and core damage frequencies using a Monte Carlo approach.

1.3.4 Fire Risk Assessment Methodology

Based on nuclear power plant operating experience over the last 20 years, it has been observed that typical nuclear power plants will have three to four significant fires over their operating lifetime. Previous probabilistic risk assessments (PRAs) have shown that fires are a significant contributor to the overall core damage frequency, contributing anywhere from 7 percent to 50 percent of the total (considering contributions from internal, seismic, flood, fire, and other events). Because of the relatively high core damage contribution, fires need to be examined in more detail.

An overview of the simplified fire PRA methodology is as follows:

Step a - Initial Plant Visit

Based on the internal event and seismic analyses, the general location of cables and components of the systems of interest is known. A plant visit provides the analyst with a means of seeing the physical arrangements in each of these areas. The analyst has a fire area checklist which aids in the screening analysis and in the quantification step.

The second purpose of the initial plant visit is to confirm with plant personnel that the documentation being used is in fact the best available information and to get clarification about any questions that might have arisen in a review of the documentation. Also, a thorough review of firefighting procedures is conducted.

Step b - Screening

It is necessary to select important fire locations within the power plant under investigation having the greatest potential for producing risk-dominant accident sequences. The objectives of location selection are somewhat competing and should be balanced in a meaningful risk assessment study. The first objective is to maximize the possibility that all important locations are analyzed, and this leads to the consideration of a potentially large number of candidate locations. The second objective is to minimize the effort spent in the quantification of event trees and fault trees for fire locations that turn out to be unimportant. A proper balance of these objectives is one that results in an optimum allocation of resources and efficiency of assessment.

The screening analysis is comprised of:

- a. Identification of relevant fire areas
- b. Screening fire areas on probable fire-induced initiating events
- c. Screening fire areas on both order and frequency of cut sets
- d. Numerical evaluation and culling on probability for each remaining fire area

Step c - Quantification

After the screening analysis has eliminated all but the probabilistically-significant fire areas, quantification of dominant cut sets is completed as follows:

- a. Determine temperature response in each fire area
- b. Compute component fire fragilities
- c. Assess the probability of barrier failure for all remaining combinations of fire areas
- d. Perform a recovery analysis

Finally, an uncertainty analysis is performed to estimate error bounds on the computed fire-induced core damage frequencies. The N-reactor fire analysis is presented in Chapter 5.

1.4 References

1. NRC Task Action Plan (Rev. 2), Shutdown Decay Heat Removal Requirements (Task A-45), draft program description issue by Generic Issues Branch, U.S. Nuclear Regulatory Commission.
2. M. P. Bohn, and J. A. Lambright Recommended Procedures for Simplified External Event Risk Analyses, (draft) Sandia National Laboratories, Albuquerque, NM, NUREG/CR-4840 February, 1988.
3. U.S. Nuclear Regulatory Commission, Reactor Risk Reference Document, NUREG-1150, Vol. 1, February 1987.
4. U.S. Nuclear Regulatory Commission, Reactor Safety Study: An Assessment of Accident Risks in U.S. Nuclear Power Plants, WASH-1400, NUREG-75/014, 1975.
5. Zentner, M. D. et. al., 1988, "N Reactor Level 1 Probabilistic Risk Assessment: Final Report," WHC-EP-0322, Westinghouse Hanford Company, Richland, Washington.
6. U.S. Nuclear Regulatory Commission, PRA Procedures Guide, NUREG/CR-2300, January 1983.
7. Kennedy, R. P. Blejwai, T. E., and D. E. Bennett, Capacity of Nuclear Power Plant Structures to Resist Blast Loadings, USNRC, NUREG/CR-2462, September 1983.
8. M. K. Ravindra, and H. Banon, Scoping Quantification of External Events in PRA of Nuclear Power Plants, NTS Structural Mechanics Associates draft report NTS/SMA 12605.01-R2, February 1985.
9. M. P. Bohn et al., Application of the SSMRP Methodology to the Seismic Risk at the Zion Nuclear Power Plant, Lawrence Livermore National Laboratory, Livermore, CA, NUREG/CR-3428, 1983.
10. D. L. Bernreuter et al., Seismic Hazard Characterization of the Eastern United States: Methodology and Interim Results for Ten Sites, NUREG/CR-3756, April 1984.
11. R. B. Kulkarni, et al., Evaluation of Seismic Exposure for the N-Reactor Plant, Hanford, WA, Report prepared by Woodward-Clyde Consultants, published as UNC Nuclear Industries, Inc. report UNI 4426, May 1987.
12. U.S. Army Corps of Engineers, Subsystem Hardness Assurance Report, HNDDSP-72-156-ED-R, Vols I and II, 1975.
13. P. B. Schnabel, J. Lysmer, and H. B. Seed, SHAKE--A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites, Earthquake Engineering Research Center, University of California, Berkeley, CA, EERC 72-12, 1972.

14. H. L. Wong and J. E. Luco, Soil-Structure Interaction: A Linear Continuum Mechanics Approach (CLASSI), Dept. of Civil Engineering, University of Southern California, Los Angeles, CA, CE79-03, 1980.
15. R. B. Worrell, SETS Reference Manual, Sandia National Laboratories, Albuquerque, NM, SAND83-2675, NUREG/CR-4213, May 1985.

2.0 PLANT DESCRIPTION

2.1 Description of Plant Systems

2.1.1 Introduction

This chapter discusses the system models of the four frontline systems that were identified as important to safety. System models discussed in this chapter were developed as part of the N-Reactor Level 1 PRA (Ref. 1). This chapter and supporting documentation in plant system notebooks provide a complete description of the facility operating systems. Complete fault tree models were identified for the following frontline systems:

- a. Reactor trip system (RTS)
- b. Primary/secondary cooling systems (PCS/SCS)
- c. Emergency core cooling system (ECCS)
- d. Graphite and shield cooling system (GSCS)

The support systems that were identified as important to safety are those that provide necessary support functions for the four frontline systems. These include high-pressure injection (HPI), circulating raw water (CRW), low-pressure raw water (LPRW or RWS-1), high-pressure raw water (HPRW or RWS-2), service water backup, demineralized water (DW), filtered water (FW), AC power, DC power, plant air, and diesel starting air. Complete fault-tree models were also developed by the internal events analysts for these support systems.

The discussion of the system analyses for the frontline systems and major support systems includes:

- a. A brief functional description of the system with reference to the one-line diagrams that were developed to indicate which components were included in the model
- b. Safety-related success criteria that were applied to the system
- c. Interfaces between the frontline systems and the support systems

2.1.2 Reactor Trip System (RTS)

The RTS is a protective system that acts to limit the consequences of an abnormal condition by automatically shutting down the reactor. The RTS consists of the safety circuits and hardware for the operation of the Horizontal Control Rods (HCRs) and the Boron Ball Safety System (BBSS), a backup safety feature to protect against a failure of the HCRs. The safety circuit includes all electronic equipment, signal processing equipment, and cables from the system sensors to the HCR hydraulic system and to the latching mechanisms on the ball hoppers.

The HCRs are hydraulically driven in and out of the reactor. Each of the 84 control rods is attached to a hydraulic cylinder that is driven under

scram conditions by hydraulic pressure from an accumulator vessel supplied from a high- and low-volume hydraulic pump. The RTS first causes the shutdown rods to be inserted into the reactor (if any of the critical parameters should indicate an out-of-limit condition), which then causes a BBSS drop into the reactor (if the rods fail to insert in time). Either the HCR system or the BBSS have sufficient capacity to preclude the possibility of subsequent recriticality.

For the purpose of this analysis, the postscram discontinuous action signals (DA) have been included in the RTS system notebook. Although not a part of the reactor trip protection system, they directly interface with the safety circuit to provide various postscram system responses. The DAs are primarily designed to do the following:

- a. Initiate a reduction in primary system flow and cooling to prevent overcooling of the reactor, ensure an adequate steam supply for operation of the primary coolant pumps (PCP), and reduce thermal shock to the primary piping
- b. Present the loss of PCS inventory by reducing shrinkage and maximizing coolant makeup additions to ensure that recirculation cooling ability is not lost
- c. Minimize the transient that occurs as inventory is regained in the PCS.

The success of the RTS is analyzed for accidents at full-power operation only. The RTS is successful if it provides a trip signal that results in a successful response by the horizontal rod mechanical system or the ball drop mechanical system, causing the reactor to become and remain subcritical. The rod safety system is designed and operated so that two rods may fail without system degradation even under worst case reactivity insertion accidents. The BBSS is designed and operated so that two ball hoppers can be out of service and the five ball hoppers with the highest neutron absorption value can fail without degrading the system.

The success for the DA circuits is achieved if the DA provides an actuation signal to its respective system after a scram.

The RTS interfaces with the following systems:

- a. ECCS--starts the diesel-driven ECCS pumps on a reactor scram
- b. Electric power system (EPS)--The reactor trip system uses 125-V DC and 120-V AC power from onsite or offsite sources
- c. Annunciator system provides indication of various abnormalities for all reactor trip signal inputs

The DA forms a reactor trip interface with the following systems:

- a. ESFAS--The RTS is activated by the Engineering Safeguards Features Actuation System (ESFAS) signal

- b. PCS--The RTS provides the signal to reset the steam valves on the PC pump drive turbines (DT) for reduction to 30 percent primary coolant flow and turns off all pressurizer heaters
- c. HPI system--The reactor trip signal starts the standby HPIPs
- d. AHR system--Starts the standby AHR pumps
- e. Plant service boiler (PSB) system--The reactor trip signal starts the feed pumps for the standby service boiler
- f. Reduces PCS flow through the steam generators by allowing one third of the flow to bypass the steam generators

The EPS is the only required support system for the RTS. However, the RTS does not require AC or DC for effective responses to accident demands. The reactor will trip in the event of loss of 125-V DC or 120-V AC.

2.1.3 Primary/Secondary Cooling System (PCS/SCS)

The PCS and SCS act together as the principal means of removing heat from the reactor. The PCS is required to remove decay heat from the reactor fuel elements. The SCS removes heat from the PCS by heat transfer in the steam generators. The steam produced by this heat transfer and boiling of condensate in the steam generators is used to supply the PCS pump-DTs, B Bus turbine generator (TG), and Supply System TGs. After a scram, during the steam depressurization state, boiler steam is brought online to supply the PCS DTs and the onsite B Bus TG. The ultimate heat sink for nuclear steam is the 109-N dump condensers (DC).

The PCS is a single closed loop that divides into five cells (plus one inactive cell). Each cell contains one PCS pump and two steam generators. The PCS pumps are normally driven by steam turbines, but can also be driven by electrically powered pony motors. The loop also contains a pressurizer for pressure control and divides through a set of headers at the reactor, into 1003 pressure tubes. Two HPIPs normally supply DW as makeup. A simplified diagram of the PCS is shown in Figure 2.1.

The SCS is also a closed-loop system. It consists principally of ten steam generators (two per PCS pump) and associated auxiliary equipment that supplies steam to various turbines (Supply System and PCS DTs) and/or DCs. Steam can also be supplied from one of two boiler sets. Condensate is returned from the steam users by some combination of 10 DT condensate pumps, two TG condensate pumps, 16 DC condensate pumps, and 4 Supply System feedwater pumps. Other principal components of the SCS include 16 DCs for excess heat removal and 3 surge tanks for inventory and pressure control. The DW is normally supplied to the SCS from the deaerator storage tank, but emergency makeup to the SCS can be supplied from the AHR system. A simplified diagram of the SCS is shown in Figure 2.2.

Figure 2.1 Primary Cooling System Diagram

Figure 2.2 Secondary System One-Line Diagram

The PCS/SCS can be operated in at least two different shutdown modes. Immediately after a scram the PCS/SCS is operated in the steaming mode. In this mode, steam is produced in the steam generators and sent to the after-scram heat sinks. These heat sinks include the TG, the DTs, and DCs, but not the Supply System TGs. Normally, approximately 3 h after a scram the PCS/SCS are used as heat exchangers without generating steam (the amount of decay heat produced by the reactor has decreased significantly by this time). The DCs are the ultimate heat sink for the reactor decay heat when operating in this mode.

The PCS/SCS must perform two functions to be considered successful. First, it must be able to remove enough heat from the core to prevent any fuel damage. Second, it has to perform in a way that the ECCS is not automatically actuated. In all cases the temperature, pressure, and flow requirements that are needed to stay away from the ECCS trip points are more restrictive than the thermodynamic requirements to cool the reactor core and prevent fuel damage. The after-scram success criteria for the PCS/SCS is as follows:

- a. Two out of five steam-driven primary cooling pumps are operating (one on each side of the reactor), or three out of five pony motor-driven primary cooling pumps. The RTS provides the signal to reset the steam valves on the PC pump drive turbines (DT) for reduction in PC pump speed from 3600 rpm to 1200 rpm.
- b. At least one steam generator must be in service to the respective operating PCP.
- c. The DCs are the ultimate secondary side heat sink which require one out of 16 DCs in service and two out of 16 DC condensate pumps operating.

Failure of these systems that support the PCS/SCS can also fail the PCS/SCS:

- a. HPI system
- b. DW system
- c. AC power system
- d. Plant air system
- e. Raw water system (RWS)

The PCS/SCS interfaces with three other major systems, the ECCS, HPI system, and RTS. The ECCS flowpath has piping that is common with the PCS, beginning at the inlet risers and ending at the 10-in. high-pressure flush manifold on the outlet side of the reactor.

The HPI system supports the PCS in a number of important functions. It provides PCS makeup, PCS pressurizer spray makeup, and seal water to the PCS pumps. Each of these functions is vital to normal full-power operation.

The RTS monitors various primary and secondary system parameters that are critical to nuclear safety. When a parameter reaches a predetermined setpoint, a reactor scram is initiated.

2.1.4 Emergency Core Cooling System (ECCS)

The ECCS, an engineered safety system, provides a separate, independent water source for emergency once-through cooling of the reactor. Three diesel-driven, high-lift (high-pressure) pumps (HLDP) are provided to deliver the cooling water to the bottom of the core inlet risers. The water cannot enter the reactor until the V-3 valves that are located upstream of the inlet risers are opened. The ECCS water then enters the PCS through check valves when the system pressure falls below approximately 300 lbf/in². The V-4 valves at the outlet of the reactor must open to depressurize the reactor and allow for the once-through cooling flow. The high-lift pumps are initially supplied by treated water from the DWT, which is maintained with a minimum reserve of 320,000 gallons before ECCS bypass trips intended for ECCS use only. After this supply is depleted, river water is automatically supplied from three low-lift, low-pressure, diesel-driven pumps (LLDP) located in the river pump house (181-N). Filtered water from the FW tank may be used instead of river water, but this requires manual action to line up the proper piping. Filtered water is preferred to river water if the piping realignment can be performed in time. The ECCS is intended for use only if the PCS/SCS is disabled.

The ECCS consists of the following major components:

- a. One DWT
- b. Optional filtered water tank (FWT)
- c. Three low-lift, diesel-driven pumps
- d. A bypass system that diverts untreated river water from the low-lift pumps back to the river until the DW and FW supplies have been used up, and then channels the river water to the high-lift diesel pumps suction
- e. Three HLDPs
- f. An ECCS pipe and manifold that delivers emergency coolant from 182-N to the reactor in 105-N
- g. Eight automatic inlet valves (V-3)
- h. Eight automatic dump valves (V-4)

- i. An emergency dump tank for quenching ECCS flow
- j. Radioactive waste disposal basin
- k. Instrumentation to actuate the ECCS
- l. Independent stored energy systems (DC batteries, compressed air accumulators, diesel fuel oil)

The one-line diagram for the ECCS is shown in Figure 2.3.

The ECCS, which is in standby during power operation, must be able to supply emergency cooling flow to the reactor pressure tubes if the normal PCS has failed. When required, the ECCS must function until the GSCS is capable of removing decay heat. If the GSCS is not available, the ECCS must continue to function. The success criteria for ECCS for all initiators are:

- a. Two out of three low-lift diesel engines must start and run
- b. Two out of four ECCS reactor inlet valves (V-3s) on each of two banks must open if all high-lift diesel engines start and run
- c. Three out of four ECCS reactor inlet valves (V-3s) on each of two banks must open if one high-lift diesel engine fails to start and run
- d. One out of four ECCS reactor outlet valves (V-4s) on each of the two banks must open
- e. All flow paths, from the last inlet header to the reactor via the inlet risers (from V-1 valves to V-11 valves), must be open

The major system-to-system interface is between the PCS and the ECCS. Since the ECCS takes suction from the DW storage tank and can take suction from the FW tank, these two water systems also interface with it. The high-lift diesel pumps recirculate to the emergency raw water (ERW) tank during HLDP functional testing, forming another potential interface, but the valve providing this interface is normally closed. Its position is included on the startup checklist as well as being annunciated in the control room. Finally, on a loss of power and a need of confinement spray, the LLDPs supply suction to the fog-spray system. The RTS automatically starts the diesel pumps. The ESFAS automatically opens the V-3 and V-4 valves and can also start the pumps. Other potential interfaces with water sources for the HLDPs have not been included in the analysis, but could be considered as recovery actions, if necessary.

The ECCS has redundant active components. Actuation and operation of the ECCS are independent of both offsite and onsite AC power sources. A multitude of redundant trip circuits ensure actuation. Redundant and independent DC battery systems, compressed air accumulators, and stored diesel fuel ensure prompt starting and operation of the ECCS.

Figure 2.3 Emergency Core Cooling System Diagram

Power to the ECCS control circuits 1 and 2 is from 125-V DC power panels CD (A battery) and CN (B battery), respectively. This provides redundant power supplies to the circuits.

The ECCS is designed with two independent, but identical, instrument trip circuits. The AC system charges the DC battery supply, and if AC power is lost, DC batteries supply emergency electricity to the instrument circuit. To prevent spurious actuation, the ECCS is designed as a fail-negative system. That is, it takes the proper amount and combination of trip functions to energize the ECCS master relay that in turn opens the V-3 and V-4 valves. These valves require DC power and air to open. Either of the two trip circuits are actuated by any of the following:

- a. Manual pushbutton
- b. Two of four confiner high-pressure trips coincident with two of three low primary coolant pressure trips
- c. Any combination of two of the following:
 - Two of four pressure tube low flow trips
 - Two of the primary coolant high-outlet-temperature trips
 - Two of three high or two of three low-primary-coolant-pressure trips
 - Four of six primary pump low shaft speed

The low-lift pumps and delivery piping up to the ECCS bypass are used for the operation of the emergency backup cooling systems for the confinement fog-spray system, the GSCS, the HCR cooling system, and the fire system. On electric loss and fog-spray demand, the low-lift pumps supply water to all of these.

The high-pressure booster air system and local air bottles provide all the ECCS compressed air requirements for the operation of the V-3 and V-4 valves. Starting air is used to start the low-lift and high-lift diesel engine starting system.

2.1.5 Graphite and Shield Cooling System (GSCS)

The GSCS provides recirculating cooling for the reactor graphite moderator, reactor thermal shields, and primary shields to control the temperature of these components during reactor operation. Two hours after a reactor shutdown, and with no other core cooling available, the GSCS provides sufficient heat removal capability to prevent fuel damage. Only the decay heat removal capability for the graphite moderator is included in this analysis. Other GSCS functions, although significant in terms of long-term structural integrity, are not significant for short-term moderator cooling in response to accident conditions.

The normal GSCS recirculation mode includes two of three operating recirculation pumps, three of four in-service heat exchangers cooled by the CRW, normal system makeup via the low-pressure injection (LPI) system, and the GSCS surge tank for inventory and pressure control. Recirculation pump-seal water is also provided by the LPI system.

An emergency once-through dump cooling mode is also available with supply from the RWS-2, which includes two independent sources of cooling water, the HPRW pumps and the FSDPs. These backup cooling water sources are analyzed as part of the RWS. Switchover to once-through operation is initiated by opening redundant supply valves from the HPRW system and redundant discharge valves to the radioactive drain system that discharges the effluent to the Liquid Effluent Retention Facility (LERF). The one-line diagram of the GSCS is shown in Figure 2.4.

The GSCS must be available to remove at least 45 MW thermal from the reactor core during power operation and shutdown until graphite temperatures are less than or equal to 200°F. In the event that all direct cooling of the fuel is lost, the GSCS provides cooling to limit fuel damage. The period of time required for decay heat removal before the GSCS can provide sufficient cooling capability to prevent any fuel damage is assumed to be 2 hours for this study. The success criteria for this mode of operation are:

- a. Operation in the recirculation mode with two of three recirculation pumps, three of four heat exchangers, adequate DW makeup, surge tank, and CRW supply to the heat exchangers
- b. Operation in the emergency once-through cooling mode with raw water supply from RWS-2

In addition to the automatic transfer to single-pass cooling initiation ("dump"), a manual transfer capability is available. Both the RWS-2 supply valves and the dump valves must be in service. If a dump occurs for any reason, the reactor must be shutdown. The GSCS transfer from recirculation to single-pass cooling will occur automatically if:

- a. Two of three GSCS coolant bulk outlet temperatures exceed 250°F for more than 20 seconds
- b. GSCS coolant bulk flow drops to 2000 gal/min or less for more than 0.5 seconds

If either of these limits is exceeded without an automatic dump, a manual dump must be initiated and the reactor must be shutdown.

The GSCS provides a backup heat removal capability to the PCS by removing up to 60 MW thermal from the reactor core. If the GSCS cooling fails, the PCS would be required to provide additional heat-removal capability. Loss of all modes of GSCS cooling leads to a small reactivity insertion. This would usually cause a rate-of-rise trip leading to SCRAM. Overall, N-Reactor is an undermoderated reactor; therefore, a loss of coolant or a temperature-related density decrease will cause a reduction in reactivity.

Figure 2.4 Graphite and Shield Cooling System Diagram

The instrumentation for the normal GSCS controls is powered by 120-V AC power with automatic DC power backup. This automatic backup function is included in the DC power fault tree. The GSCS dump initiation circuitry is powered by 125-V DC power (105-N DC systems) and will transfer to the once-through cooling mode on loss of power.

2.1.6 Raw Water Systems (RWS)

The RWS indirectly supplies makeup to the closed-cycle cooling systems (PCS and GSCS) and provides the ultimate heat sink for the secondary system via the DCs, surface condensers, etc. The water is initially supplied from the Columbia River through the CRW pumps at the river pumphouse (181-N).

Four separate systems are included in this analysis:

- a. Circulating Raw Water (CRW)
- b. Low-Pressure Raw Water (RWS-1)
- c. High-Pressure Raw Water (RWS-2)
- d. Service Water Backup System.

Functional descriptions of these systems are provided in the following subsections.

2.1.6.1 The CRW system (shown in Figure 2.5) is the primary source of all water used in the plant and interfaces with all other water systems.

The CRW system originates at the river pump house (181-N). It consists of four CRW pumps that supply Columbia River water for a variety of plant safety-related uses, including:

- a. Shell side of GSCS heat exchangers
- b. PCS DT condensers and oil coolers
- c. Tube side of 16 SCS DCs
- d. Service water backup pump supply
- e. Supply header for RWS-1 and RWS-2 pump supply
- f. TG condenser cooling.

The CRW system success criteria is as follows:

- a. Three of four CRW pumps operate during normal reactor operation (two on B Bus). One pump must continue to operate during any transient until cold shutdown is reached

Figure 2.5 Circulating Raw Water Diagram

- b. For all nonrunning CRW pumps, either the discharge check valve (RWSV-806's) or the discharge hydraulic valve (RWSV-805's) must be operable or, if the valve has failed, it must be in the closed position

2.1.6.2 The low-pressure raw water system (shown in Figure 2.6) takes water from the CRW system and provides single-pass cooling water to the HPI lube oil and fluid coupling coolers. Other related uses of RWS-1 water include auxiliary cooling supply for HLDP and fog-spray diesel pump (FSDP) jacket cooling. The RWS-1 success criteria is that one of three pumps continue to operate.

2.1.6.3 The high-pressure raw system (shown in Figure 2.7) consists of two sets of pumps: HPRW pumps and FSDPs. These pumps are supplied with water from the CRW pumps. In addition, the FSDP can be supplied by the low-lift diesel pumps (LLDPs). The RWS-2 System supplies water for the once-through cooling mode of GSCS and confinement fog sprays for 105-N and 109-N.

The RWS-2 success criteria is as follows:

- a. Two of three RWS-2 pumps continue to operate
- b. One of two FSDPs starts on demand and continues to operate.

2.1.6.4 The service water backup system (included in Figure 2.5, CRW system) backs up the normal service water that is supplied by the low-pressure filtered water (LPFW) system. This system provides cooling water for:

- a. TG hydrogen coolers
- b. Boiler feed pump coolers
- c. 184-N air compressor and aftercoolers.

The success criteria is that two of three pumps start and continue to run to backup the LPFW service water system.

2.1.7 High-Pressure Injection System (HPI)

The HPI system provides DW to the PCS as coolant makeup during normal operation. The HPIPs take their suction from the DWT and supply the following:

- a. Primary loop makeup
- b. PCP seal water
- c. Primary loop cold sprays.

In accident or transient situations, such as a small LOCA or during a scram, the HPI system provides the necessary extra makeup. (A small LOCA

Figure 2.6 Low Pressure Raw Water Diagram

is a break that is small enough that the HPI makeup will be sufficient to prevent the need for ECCS actuation.)

The HPI system is composed of five pumps driven by AC motors, the DWT supply, and the associated valves and piping. The one-line diagram of the HPI system is shown in Figure 2.8.

Normally the HPI system makes up 800 to 1000 gal/min to the primary system with two pumps operating (one on each bus). Normal pump discharge pressure is approximately 1700 lbf/in² but the system is capable of producing more than 2000 lbf/in² if required. In the event of a scram the standby HPIPs start automatically to make up for any inventory losses from the primary system.

The HPI system normally has four operable pumps; two operating (one on each bus) and two in automatic standby (one on each bus). The success criteria state that, initially, there are four operable pumps; two are running and two are in automatic standby. For a small LOCA initiator, at least one additional pump must start and continue to run for a three-out-of-four success criteria. For a transient initiator, a two-out-of-four success criteria is sufficient.

The HPI system provides normal makeup to the PCS seal water and the PCPs, and supplies the primary pressurizer cold sprays used for pressure control of the PCS.

The HPIP fluid drive couplings are controlled using the control air system, with diesel starting air as a backup. The pump recirculation valves also use control air.

The HPIP fluid drive couplings and oil system are cooled by the LPRW system.

2.1.8 Demineralized Water System (DW)

The equipment that comprises the DW system includes the DWT, AHR tank, LPI pumps, AHR pumps, SLMU pumps, and associated valves and piping. The DW is stored in the DWT and AHR tank until it is distributed for use throughout the plant via the LPI, AHR, SLMU, and HPIPs. The HPI system is modeled separately as an individual system and therefore is not included as part of the DW system. The one-line diagram of the DW system is shown in Figure 2.9. Safety functions of the DW system include suction to HPIPs and water supply for ECCS cooling for 20 minutes.

The DW system provides water for many plant systems. It must be available to supply cooling water during reactor operation and shutdown. The success criteria include:

- a. Two out of three LPI pumps operating
- b. One out of three AHR pumps operable
- c. Two out of two SLMU pumps operating

Figure 2.9 Demineralized Water System Diagram

- d. At least 75 percent water level in the DWT
- e. At least 25 percent water level in the AHR tank

The DW system interfaces with the FW system where the LPFW pumps provide the demineralized plant supply. It also interfaces with the following plant systems that it supplies:

- a. Primary system makeup, and PCP cooling
- b. Secondary system makeup, and DC coolant pump (DCCP) cooling
- c. GSCS makeup, and GSCS pump cooling
- d. Boiler system makeup
- e. Rod cooling system makeup, and rod coolant pump cooling

2.1.9 Filtered Water System (FW)

The LPFW and high-pressure filtered water (HPFW) pumping systems supply water to various cooling systems of the plant, including:

- a. Demineralization plant feed
- b. 117-N filter sprays
- c. Fuel storage basin
- d. Diesel jacket cooling
- e. Rod cooling

The water is supplied from the filtered water tank.

Functional descriptions of the LPFW and HPFW systems are provided below.

- a. The LPFW system (Figure 2.10) takes water from the FW tank and provides makeup water to the demineralization plant, emergency diesel jacket coolers, 117-N confinement filter sprays, and irradiated fuel storage basin
- b. The HPFW system (also included in Figure 2.10) takes water from the LPFW system and provides water to the rod cooling system.

The FW system provides water for normal N-Reactor operation in quantities sufficient to meet demands during both normal and emergency conditions.

To meet these requirements, the FWS success criteria include:

- a. Two out of four LPFW pumps operating, one on each bus

Figure 2.10 Filtered Water Systems Diagram

- b. Two out of three HPFW pumps operating, one on each bus
- c. At least 25 percent water level in the FW tank

The FW systems supply water to many plant systems, and are supported by onsite and offsite electrical power.

2.1.10 Alternating Current Power System (AC)

N-Reactor uses two independent sources of AC power designated as A and B Bus, both which must be in service when the reactor is operating. The A Bus is powered by offsite sources while B Bus is powered by an onsite steam-driven TG that is always running. The design philosophy for the N-Reactor is that all important loads, instrumentation, and equipment needed to deal with transients, small LOCAs, and normal operating conditions can be supplied by either A Bus or B Bus. Important systems normally have redundant pumps, half on one bus and half on the other. However, the loads supplied by these AC power sources are not required for safe shutdown with ECCS, since ECCS pumps are driven directly by diesel engines. Simplified drawings of AC power are shown in Figures 2.11 and 2.12.

The Hanford Site 230 kV loop supplies A Bus at the 13.8 kV level. This source provides redundant parallel transmission lines and transformers.

The onsite steam-driven TG is the primary power source for B Bus, although B Bus can be supplied from offsite power in emergencies. It normally receives its steam from the steam generators (nuclear steam). However, it can also be supplied by boiler steam from oil-fired boilers that are required to be on standby, ready for service.

The success criteria for A Bus and B Bus are independent of each other, because the power sources and the distribution systems are independent of each other. The success criteria follow.

- a. Within the A Bus distribution system, each bus must be available to the electrical power users at the appropriate voltage and power level
- b. Within the B Bus distribution system, each bus must be available to the electrical power users at the appropriate voltage and power level. Steam to drive the B Bus TG must be available following a scram; nuclear steam for 0 to 30 minutes and boiler steam after 30 minutes

The AC power system interfaces with almost every system in the plant by providing electrical power to those systems. The AC power system also depends on other systems for support of the TG. This includes raw water to cool the TG surface condensers and the 153-N B battery to power the emergency bearing and seal pump.

Figure 2.11 Simplified Alternating Current Power Diagram

2.1.11 Direct Current Power System (DC)

The DC power system provides a continuous source of DC power to various pieces of equipment throughout N Reactor. This system is normally supplied from offsite and onsite 480-V AC buses for all battery systems, except the Leeds and Northrup (L&N) uninterruptible power supply (UPS) battery system. The L&N UPS battery system is supplied from offsite and onsite 4160-V AC buses to the 208-V AC instrument power buses. The AC buses feed rectifiers/chargers that convert the AC to DC power and charge the associated batteries. The equipment to convert AC to DC power will be referred to as a rectifier throughout the rest of this section. Upon loss of an AC feed to a rectifier, the battery associated with that rectifier will provide the necessary DC power. It is noted that the L&N UPS DC power is converted to 120-V AC power through an inverter before it powers the L&N system. The DC power sources are independent of one another, and are found in several buildings onsite. The batteries (125-, 130-, and 24-V DC) modeled for this study are:

- a. 105-N, batteries A and B
- b. 109-N, batteries A and B
- c. 151-N, battery A
- d. 153-N, batteries A and B
- e. 181-N, battery A
- f. 182-N, battery A
- g. 105-N, batteries D and E (24-V DC)
- h. 105-N, L&N UPS battery (130-V DC)

Each DC power bus consists of a battery attached to the distribution bus and an AC/DC rectifier. Each rectifier is designed to carry any normal DC bus load, and keeps the battery fully charged and available for service. A simplified drawing of the DC system is shown in Figure 2.13 and it includes any interfacing support systems and all components modeled in the fault tree, except instrumentation and controls.

Each DC bus, except one of the two 24-V DC buses, is required to supply designated loads with a continuous source of 125-V DC, 130-V DC, or 24-V DC power. To accomplish this, each DC bus must receive voltage from either the appropriate 480-V AC bus (or 4160/208-V AC bus for L&N UPS) through the rectifier or from its battery.

Figure 2.13 Simplified Direct Current Power Diagram

The DC system provides power to various components and systems throughout the plant. It is impractical to list all equipment dependent on DC power here; however, each individual system description provides a list of components requiring DC power.

The DC system does depend on other support systems. These requirements are as follows:

- a. Buses A and B, 480-V AC, are required to supply AC power to the rectifier for conversion from AC to DC for all DC systems except the L&N UPS system. If the AC power is lost, the batteries will provide DC power for a limited amount of time
- b. Buses A and B, 4160/208-V AC, are required to supply AC power for the L&N UPS system. If the AC power is lost, the L&N battery will provide DC power for a limited amount of time

2.1.12 Plant Air System

The instrument and service air systems for N-Reactor provide compressed air to a number of reactor components and auxiliary components that are vital to both safe reactor operation and shutdown.

The instrument air system provides compressed air for the following:

- a. Control valve positioning
- b. Instrumentation

The instrument air that is used for control valve positioning and instrumentation flows from the dryers to the 184-N, 109-N, 163-N, 183-N, 182-N, and 181-N buildings. The instrument air that is used for breathing air in the areas of high airborne contamination flows from the instrument air receiver directly to the 105-N and 109-N buildings without passing through the dryers. In the 105-N building breathing air also supplies instrument air by flowing through air dryers in 105-N Room 2 to become instrument air.

The service air system provides compressed air for the following:

- a. Confinement door inflatable seals in 105-N and 109-N
- b. Ignitors for the CE oil-fired boilers in 184-N
- c. Building steam vent backup closures in 105-N
- d. Fog-spray control valves in 105-N

Each system provides limited backup for the other system by means of the following design features:

- a. Service air is automatically backed up by instrument air if the supply air pressure drops to 90 lbf/in². This is accomplished by a pressure-sensing valve (CASV-106-1) opening and allowing undried instrument air to flow into the service air line via the system's backup crosstie line
- b. Instrument air is automatically backed up by service air if the instrument air pressure drops to 90 lbf/in². This is accomplished by a pressure-sensing valve (CASV-108-1) opening and allowing service air to flow into the instrument air line upstream of the instrument air dryers
- c. There is also a pressure valve (CASV-185) that will open when the instrument air drops to 95 lbf/in² in case an air dryer clogs up or there is a valving error. The CASV-185 ensures that instrument air is supplied by bypassing the air dryers
- d. The portable diesel-driven compressor can be manually started, via a self-contained battery, to increase service air volume if the service air compressor fails or is out of service for maintenance
- e. Valve SAV-103-1 shuts off service air to nonvital functions (functions other than backup instrument air and service air to the boilers) if the total demand for compressed air exceeds the compressor or instrument air bypass capabilities

The instrument and service air drawings are shown in Figures 2.14 and 2.15. To provide adequate air for the instrument air loads, two compressors must be available. This can consist of both instrument air compressors, or a combination of one instrument air compressor and the service air compressor.

The service air loads require that one air compressor be functional to ensure an adequate air supply to its loads. This can be either the service air compressor or the two instrument air compressors. The instrument air/service air supply system provides compressed air to various components and systems throughout the plant.

2.1.13 Diesel Starting Air System

The diesel engines that power the high-lift, low-lift, and fog-spray pumps are started with a high-pressure compressed air system known as the diesel starting air system. The function of the diesel starting air system is to provide 600-lbf/in² air to start the engines and provide sufficient energy to bring them up to operating speed. The diesel starting air system drawing is shown in Figure 2.16.

The starting air receivers must be able to provide 475 lbf/in² (gage) of starting air to all of the ECCS diesels. The starting air receivers are capable of this without depending on the starting air compressors, so starting air compressor paths have been modeled only as a source of air in case a receiver leaks.

Figure 2.15 Service Air System Diagram

The diesel starting air system is designed to provide air to the diesel pumps for actuation. Backup air to the HPIPs is also provided from starting air receiver No. 3. The diesel pumps supported are listed below:

- a. High-lift diesel pumps
- b. Low-lift diesel pumps
- c. FSDPs

2.2 Accident Sequence Event Trees

2.2.1 Introduction

Three classes of accident initiators and five classes of applicable mitigating functions were established as was done in the internal event analysis. The initiator classes were:

- a. Large LOCAs
- b. Small LOCAs
- c. Transients

Large LOCAs are major pipe breaks that do not necessarily require using the V-4 valves to depressurize the reactor to allow ECCS water to enter the core. Medium LOCAs are those which always require the V-4 valves. Small LOCAs were defined as pipe breaks that could be mitigated by the high-pressure injection system (HPI) and therefore would not require an ECCS actuation. Transients were defined as off-normal events (including undercooling and over-power initiators) that would result in plant shutdown, either automatic or manual.

During the second phase of the internal events analysis a more thorough review of potential pipe breaks and plant response was completed. An analysis of primary system piping showed that only a very small class of large LOCAs would not require the V-4s to depressurize the reactor. This class consists of the 105-N Building outlet headers and the 109-N cell inlet lines. The frequency of a pipe break in these runs is small compared to the overall large LOCA frequency. Since the requirement for V-4 valves opening is more conservative, it was decided to fold medium LOCAs together with large LOCAs and have only one class of initiator. Therefore, large LOCAs are defined as any major pipe break that requires the ECCS. Accordingly, three event trees were developed, one for each class of initiating events.

The transient initiating events were included in the fault-tree models. Consequently, the effects of the initiators are contained in the fault trees and, as a result, one transient event tree adequately models all of the transient initiator groupings.

The five mitigating functions were:

- a. Reactivity control
- b. Primary coolant heat removal
- c. Secondary coolant heat removal
- d. ECCS
- e. GSCS

These functions are required for the following reasons. Primary and secondary heat removal is the normal way that decay heat is removed from the fuel, both during power operation and after the reactor is shutdown. The primary system removes heat directly from the fuel, the secondary system transfers the heat from the primary both to the environment and to generators used to power onsite and offsite loads. The ECCS provides backup heat removal capability in the event the primary/secondary heat removal capability is lost. The GSCS is designed to cool the graphite moderator, but does have the capability to remove enough heat to limit or prevent fuel damage in the event that both the primary/secondary and ECCS heat removal functions fail.

The amount of fuel damage the GSCS can prevent is dependent on the decay heat load. For the purpose of this study, it is assumed that 2 hours after a scram the GSCS acting by itself can remove enough heat to prevent any fuel damage. It is also assumed that between the time of scram and 2 hours after the time of scram the GSCS acting by itself can limit the amount of fuel damage. The event trees used in this analysis reflect these assumptions.

The following subsections describe each of the three event trees used in this study and the potential accident sequence end-states that may result from each sequence in the trees.

2.2.2 Large Loss-of-Coolant-Accident Event Tree

The initiating event for this tree (Figure 2.17) is a large pipe break, failure of major component supports or failure of Building 109, causing depressurization and coolant loss that cannot be made up by normal plant systems, resulting in a complete and prompt failure of the PCS. Mitigating systems responding to a large LOCA are:

- a. ECCS
- b. GSCS

Successful reactivity control (Event D in the tree) implies a complete and timely reactor scram on an appropriate signal.

LARGE LOCA INITIATOR	REACTIVITY CONTROL	ECCS STARTS AND FUNCTIONS	ECCS FLOW OCCURS IN ALL 16 RISERS	ECCS CONTINUES TO FUNCTION	GSCS COOLS FUEL VIA THE GRAPHITE	Endstates	Sequence Designator	Fuel Damage Sequence Number
L	D	A	A2	A1	C			
						OK	L	
						OK	LA1	L1
						Z1 (DEL)	LA1C	L2
						M	LA2	L3
						X	LA2C	L4
						M	LA2A1	L5
						Z1 (DEL)	LA2A1C	L6
						Y	LA	L7
						Z	LAC	L8
						Z	LD	

Figure 2.17 Large Loss-of-Coolant-Accident Event Tree

The ECCS function has been separated into three steps. The first step, Event A, is that the ECCS starts and functions as designed and operates for 2 hours. The assumption is that if the ECCS functions until the decay heat load has decreased to the point that the GSCS can adequately cool the core and then fails, fuel damage can be avoided.

The second step, Event A2, is that all ECCS flow paths function as required. A failure implies that flow through 1 of 16 core inlet risers has failed, probably by either a closed V-1 valve, or a CV-2R check valve failing to open. This will affect 1/16 of the core and is assumed to result in some core damage, depending on the behavior of the GSCS.

The third step, Event A1, is that the ECCS continues to function beyond 2 hours after initiation. Success in this time frame assumes that only a single high-lift and low-lift diesel are required to provide adequate cooling, as opposed to two of each for the earlier time frames.

Success for the GSCS function, Event C, assumes that the GSCS continues to function in the recirculation mode or operates successfully in the once-through mode. If the ECCS fails after 2 hours but the GSCS continues to operate, no fuel damage will occur. If the ECCS fails before the decay heat removal requirement drops to the GSCS capacity, some fuel damage will occur. A GSCS failure given early ECCS actuation but subsequent failure will likely result in delayed but complete fuel failure.

A review of the LOCA event tree reveals that there are ten large LOCA sequences with eight leading to some level of fuel damage. These eight sequences which lead to some fuel damage have been designated as L1 to L8 and are defined as follows:

- a. Sequence L1, the ECCS has started and run long enough (about 2 hours) for the GSCS to provide sufficient heat removal capability, but then both the ECCS and GSCS fail to continue to run for 24 hours.
- b. Sequence L2, all systems have been successful, but flow fails in a single riser due to valves left closed, failing to open, or failing to remain open
- c. Sequence L3 is similar to sequence L2 except that the GSCS also fails
- d. Sequence L4 involves both the riser flow failure and ECCS starting, but failing to continue to run (GSCS is successful)
- e. Sequence L5 is similar to sequence L4 but with the GSCS failed
- f. Sequence L6 represents a failure of the ECCS to start and run for 2 hours

Figure 2.18 Small Loss-of-Coolant-Accident Event Tree

- g. Sequence L7 is a combination of a failure of ECCS to start and run for 2 hours and a GSCS failure
- h. Sequence L8 is failure of the scram system only

2.2.3 Small Loss-of-Coolant-Accident Event Tree

This tree (Figure 2.18) describes accident sequences initiated by a small LOCA, defined as a small pipe break or valve failure in the primary system whose consequences can be mitigated by the primary system's normal makeup and spill control system. Mitigation includes being able to provide adequate makeup water during a reactor scram and consequent coolant shrinkage. Responding mitigating functions include:

- a. Reactivity control
- b. Normal primary/secondary heat removal
- c. ECCS
- d. GSCS

Successful reactivity control (Event D in the tree) for a small LOCA is similar to that of a large LOCA and is always assumed to occur. It is assumed that a scram demand always occurs.

Primary/secondary heat removal has been divided into two steps. The first, Event G, is that the primary/secondary systems operate to remove initial decay heat for 2 hours. This implies that the HPI system functions (three pumps operate or start), the spill and fill valves function as designed, and three of the five operating steam generator cells and reactor coolant pumps (two on one side of the reactor and one on the other) operate. The pump and cell constraints are a result of flow requirements and are not necessarily for heat removal reasons.

The second step, Event G1, is that the primary/secondary system continues to remove decay heat for at least 24 hours. A failure of the primary/secondary system will result in an ECCS initiation.

The ECCS function, Events A, A2, A1, and the GSCS function, Event C, and resulting end-states for the small LOCA are similar to those described for the large LOCA and will not be redescribed here.

A review of the small LOCA event tree reveals that there are twelve sequences that result in a fuel-damage state of some type. These sequences have been designated S1 through S10, and are defined as follows:

- a. In sequence S1, the primary/secondary system is successful after the scram for a period of less than 2 hours. The ECCS is started manually, but it fails to run for 24 hours and the GSCS also fails
- b. In sequence S1A the primary/secondary system continues to function for less than 2 hours after the scram. The ECCS starts and functions normally for 24 hours, but flow in a single riser fails and the GSCS fails to continue to function
- c. Sequence S1B is similar to sequence S1A except that the ECCS fails to continue to function for 24 hours
- d. Sequence S2 is similar to S1, but ECCS (manual start) fails immediately
- e. In sequence S3, the primary/secondary system fails immediately after the initiator. The ECCS starts and operates successfully but fails within 2 hours and the GSCS fails
- f. In sequence S4, the primary/secondary system fails as described in sequence S3. The ECCS starts and functions, but flow in a single riser fails and the GSCS functions normally
- g. Sequence S5 is similar to sequence S4, but includes failure of the GSCS to function
- h. Sequence S6 is also similar to sequence S4, with the ECCS starting and running for 2 hours but failing to continue to function for 24 hours and the the GSCS functions normally
- i. Sequence S7 is similar to sequence S6, but with failure of the GSCS
- j. In sequence S8, the primary/secondary system and the ECCS fail initially, but the GSCS is successful
- k. Sequence S9 is similar to sequence S8, but with a failed GSCS
- l. Sequence S10 is a failure of the scram system.

2.2.4 Transient Event Tree

The transient event tree (Figure 2.19) models the shutdown path the reactor will go through as a result of a scram (or a failure to scram when required) due to some off-normal event. The initiating event for this tree can be any of the internal events transient initiators as well as seismically-induced structural failures of buildings 184, 153, 151, or 182.

The transient event tree is similar in appearance to the tree used to analyze small LOCAs. However, there is one significant difference in the

Figure 2.19 Transient Event Tree

trees. This difference is in the success criteria applied to the primary/secondary system, as the number of HPIPs required is greater (three rather than two) for a small LOCA. Responding mitigating functions for the transient event tree include:

- a. Reactivity control
- b. Normal primary/secondary heat removal
- c. ECCS
- d. GSCS

Successful reactivity control (Event D in the tree) for a transient is similar to that of a large LOCA.

Primary/secondary heat removal has been divided into two steps. The first, Event G, is that the primary/secondary systems operate to remove initial decay heat. This implies that the HPI system functions (two pumps operate or start), the spill and fill valves function as designed, and three of the five operating steam generator cells and reactor coolant pumps (two on one side of the reactor and one on the other) operate. The pump and cell constraints are a result of flow requirements and are not necessarily for heat removal reasons.

The second step, Event G1, is that the primary/secondary system continues to remove decay heat. A failure of the primary/secondary system will result in an ECCS initiation.

The ECCS function, Events A, A2, and A1, and the GSCS function, Event C, and resulting end-states for transients are similar to those described for the small LOCA and will not be redescribed here.

A review of the transient event tree reveals that there are 12 sequences that result in a fuel-damage state of some type. These sequences have been designated T1 through T10 and are defined as follows:

- a. In sequence T1, the primary/secondary system is successful after the scram for a period of less than 2 hours. The ECCS is started manually, but it fails to run for 24 hours and the GSCS also fails
- b. In sequence T1A, the primary/secondary system continues to function for less than 2 hours after the scram. The ECCS starts end functions normally for 24 hours, but flow in a single riser fails. The GSCS fails to continue to function

- c. Sequence T1B is similar to sequence T1A, except that the ECCS fails to continue to function for 24 hours.
- d. Sequence T2 is similar to T1, but ECCS (manual start) fails immediately
- e. In sequence T3, the primary/secondary system fails immediately after the initiator, ECCS starts and operates successfully but fails within 2 hours and the GSCS fails
- f. In sequence T4, the primary/secondary system fails as described in sequence T3. The ECCS starts and functions, but flow in a single riser fails and the GSCS functions normally
- g. Sequence T5 is similar to sequence T4, but includes failure of the GSCS to function
- h. Sequence T6 is also similar to sequence T4, with the ECCS starting but failing to continue to function for 24 hours. The GSCS functions normally
- i. Sequence T7 is similar to sequence T6, but with failure of the GSCS
- j. In sequence T8, the primary/secondary system and the ECCS fail initially, but the GSCS is successful
- k. Sequence T9 is similar to sequence T8, but with a failed GSCS
- l. Sequence T10 is a failure of the scram system.

2.3 References

1. Zentner, M.D. et. al., 1988, "N Reactor Level 1 Probabilistic Risk Assessment: Final Report," WHC-EP-0322, Westinghouse Hanford Company, Richland, Washington.

3.0 EXTERNAL EVENTS SCOPING ANALYSIS

3.1 Summary

A scoping quantification study was performed for the N-Reactor to determine which external events should be included in the detailed PRA study. The results and recommendations of the study are summarized here.

The scoping quantification study considered all possible external events at the site except for seismic and fire events, since these two events were included in a detailed external events analysis. The PRA Procedures Guide (Ref. 1) was used as a guideline for systematic identification of all possible external events at the N-Reactor site. Next, an initial screening process was carried out to eliminate some of the events from the list. For this purpose, a set of screening criteria was developed and then each external event was examined for possible elimination based on these criteria. After the initial screening process was completed, the following events were found to be potential contributors to plant risk:

- a. Extreme winds and tornadoes
- b. External flooding
- c. Internal flooding

A bounding analysis was done for these events. The degree of sophistication in the bounding analysis for each event depended on whether the event could be eliminated based on only a hazard analysis or a complete analysis including hazard analysis, fragility evaluation and plant response analysis.

Tornadoes, extreme winds and tornado missile impacts were eliminated on the basis of low tornado strike probability and features of plant structures and components designed to withstand the effects of the design basis tornado. (Refer to Section 3.4.) Internal flooding was eliminated based on a set of screening criteria outlined in Section 3.6.

External flooding due to local precipitation or hydrological events was found to contribute negligibly to plant risk. But flooding due to random or seismically induced dam failure may be a significant contributor. This conclusion is, however, based on a number of generic assumptions, which may not reflect the actual condition of dams in the system as discussed in Section 3.5.

3.2 General Description

N-Reactor is located on the banks of the Columbia River in the Hanford Reservation, Washington, near the town of Richland. The Hanford Reservation covers a semi-arid area of 570 square miles in southeastern Washington and is controlled by the Department of Energy (Figure 3.1). It occupies part of Benton, Franklin, Adams and Grant Counties. The area is flat with gently rolling hills leading to higher mountains. The site is on a plateau that contains a variety of nuclear facilities, most of which have now been closed. The 100-N area, referred to as N-Reactor, is situated at river mile 379.5 (Figure 3.2) and is approximately 17 miles downstream of Priest Rapids Dam.

Figure 3.1 Location of Hanford Reservation

The area within 50 miles of 100-N site is primarily agricultural. Land use studies show homes, business, light industry, farming and nuclear facilities supporting a population of 340,000 people. State route 24 passes within 5-1/2 miles of the N-Reactor site. A railroad branch to the N-Reactor site exclusively serves Hanford operations.

A nearby river is used mostly for recreational purposes and flow is controlled from upstream dams. There are no military bases, missile sites, industrial or manufacturing plants, chemical plants and storage, airports or transportation facilities within the close vicinity of the site.

Hanford is in the rain shadow of the Cascade mountains and therefore, annual precipitation is only 6.28 inches, most of it in winter. Snow accumulates on the ground as much as 6 inches every one in eight winters. The diurnal range of wind is considerable and is due to topographic channeling. Winter minimum temperature varies from -27°F to +22°F and summer maximum from 100°F to 115°F. Thunderstorms occur frequently but tornadoes are rare. Atmospheric stagnation is one of the notable features of the wind regime.

The plot plan of the 100-N site is shown in Figure 3.3. The reactor and its cooling systems are located in the adjoining buildings 105-N and 109-N. Central parts of these buildings are designated as the reactor confinement, and consists of a 3-ft to 5-ft 9 in.-thick reinforced concrete structure. The roof is constructed from steel girders and roof trusses. Building 181-N is the reinforced concrete river pumphouse and 182-N houses diesel driven pumps.

3.3 Initial Screening of External Events

An extensive review of information on the site region and plant design was made to identify all external events to be considered. The data in the NUSAR (Ref. 2) as well as other data obtained from a variety of sources, including the information gathered during the site visit, was reviewed for this purpose. Table 3.1 is a listing of external hazards for the N-Reactor site based on an augmentation of Table 10.1 of the PRA Procedures Guide (Ref. 1). A set of screening criteria was developed to minimize the possibility of omitting significant risk contributors while reducing the amount of analysis to manageable proportions. These screening criteria are listed at the end of Table 3.1. For each external event, the applicable screening criteria and a brief remark is included in this table.

The findings of the preliminary screening are that, aside from seismic and fire which have already been included in the detailed event analysis, the following events were identified for a bounding study:

- a. Extreme winds and tornadoes
- b. External flooding
- c. Internal flooding

These events are discussed in Sections 3.4, 3.5, and 3.6, respectively.

Figure 3.3 N-Reactor Plot Plan

Table 3.1

Initial Screening of External Events for N-Reactor

Event	Applicable Screening Criteria*	Remarks
Aircraft Impact	3	There are no private or commercial airports within a distance of approximately 25 miles from the site
Avalanche	3	Topography is such that no avalanche is possible near this site
Biological Events	5	The only biological event which may affect the safety of the plant is fish in the river, i.e., fish may block flow of water in the intake structure. This event is considered insignificant to risk because there would be adequate warning
Coastal Erosion	3	N-Reactor site is located on the bank of the Columbia River. There are several flood control dams upstream and downstream of the plant. Therefore, erosion is not a significant concern
Dam Failure	4	Included in the effects of external flooding
Drought	1,5	The flow in the river is well regulated and there would be adequate warning so that remedial action could be taken
External Flooding	---	A bounding analysis is performed for this event
Extreme Winds and Tornadoes	---	A bounding analysis is performed for this event
Fog	3	Fog can affect the frequency of occurrence of other hazards such as highway accidents or aircraft landing and take-off accidents. None of these events are of concern for this site because of the remote location of N-Reactor from transportation network and airports

Table 3.1

Initial Screening of External Events for N-Reactor
(Continued)

<u>Event</u>	<u>Applicable Screening Criteria*</u>	<u>Remarks</u>
Forest Fire	3	Brush fires have occurred on the Hanford Reservation due to lightning. The areas adjacent to major buildings and auxiliary facilities of N-Reactor are maintained so that weed growth is prevented by landscaping, gravel ground cover and/or weed control spraying. A DOE fire station dedicated to the N-Reactor is located less than 3 miles from the N-Reactor. Therefore, this event is not of significance to the site.
Frost	1	Loads induced on structures due to frost are much lower than snow and ice loads, i.e., frost loads can be safely neglected in the plant hazard analysis.
Hail	1	Hail is less damaging than the tornado missiles. Therefore, hail is not considered further in the scoping study.
High Tide, High Lake Level or High River Stage	4,5	Included under external flooding
High Summer Temperature	3,5	As mentioned under drought, it is possible to safely shutdown the plant because of sufficient warning time. Therefore, high temperatures are indirectly included under drought conditions.
Hurricane	3	This site is well inland and further protected from the influence of the oceanic winds by high mountain ranges.
Ice Cover	1	Ice or snow loading is considered in the plant design. Ice blockage of river is included in flood.

Table 3.1

Initial Screening of External Events for N-Reactor
(Continued)

<u>Event</u>	<u>Applicable Screening Criteria*</u>	<u>Remarks</u>
Industrial or Military Facility Accident	3	There are no military bases, manufacturing plants or chemical plants within a 5-1/2 mile radius of N-Reactor
Internal Flooding	---	A bounding analysis is performed for this event
Landslide	3	The N-Reactor site is on relatively flat land where no landslides are possible
Lightning	1	Plants are usually designed for lightning
Low Lake or River Water Level	1,5	The condition should not occur because of the regulated flow in the river; also, there will be adequate warning so that remedial action could be taken.
Low Winter Temperature	1,5	Low temperatures can affect the plant structures as well as the river. Thermal stress and embrittlement which are induced by low temperatures are insignificant compared to other design loads. In addition, these effects are covered by design codes and standards for plant design. Ice cover on the Columbia River does not affect the plant safety since there is adequate warning so that remedial action can be taken
Meteorite	2	This event has a very low probability of occurrence. A study by Solomon (Ref. 3) showed that the probability of a meteorite impacting any nuclear power plant in U.S. is negligible, and therefore meteorites will not be considered in the scoping study

Table 3.1

Initial Screening of External Events for N-Reactor
(Continued)

Event	Applicable Screening Criteria*	Remarks
Pipeline Accident	3	There are no pipelines within 35 miles of the site
Intense Precipitation	4	Included under external flooding
Release of Chemicals From On-Site Storage	3,5	<p>The chemicals stored at the site include radioactive wastes from nuclear operations. These storage locations are isolated from accessible areas and appropriately marked with concrete markers and warning signs.</p> <p>Other chemicals stored on the site include fuel oil, acetone, ammonia, carbon dioxide, chlorine, helium, nitrogen and sulfuric acid. Chlorine and ammonia pose a potential risk because of their toxicity, quantity and volatility. Fuel oil (1.5 million gallons) may also pose problems due to smoke entering the control rooms during a fire.</p> <p>A control room habitability upgrade project H-544 was completed through final design as part of the most recent N-Reactor design upgrade program. If a decision is made to restart the facility, final construction and acceptance testing would be completed for these modifications. Full details of habitability improvements are described in section 6.4 of the NUSAR, (Ref. 2) amendment 21 dated September, 1989. In general, the improvements provide operator protection for anticipated accident conditions including radiological hazards and toxic chemical hazards. Local inventories of chlorine,</p>

Table 3.1

Initial Screening of External Events for N-Reactor
(Continued)

Event	Applicable Screening Criteria*	Remarks
		phosphoric acid, sodium hydroxide and ammonia hydroxide, and hydrazine which represented the largest and most proximate chemical hazards were specifically addressed in the design
River Diversion	3	This event is not relevant for the site under consideration. The river is well regulated
Sand Storm	3	This event is not of significance to the site
Seiche	3	This event is not relevant for this site
Seismic Activity	---	This event is included in the detailed external events analysis
Snow	1	Extreme snow loads are expected to be less than the design roof loads
Soil Shrink Swell Consolidation	5	Such effects occur over a long period and they do not pose a hazard during plant operation, i.e., the plant can be safely shutdown
Storm Surge	3	The site is not close enough to any large water body to experience surge flooding
Transportation Accidents	3	The nearest highway (State Route 24) is 5-1/2 miles from the N-Reactor and traffic is very light. There is no barge traffic past the site and the river is used only for recreational activities. The barge traffic terminates at Richland. An existing

Table 3.1

Initial Screening of External Events for N-Reactor
(Continued)

Event	Applicable Screening Criteria*	Remarks
		railroad has a branch line to the site, which runs within 2600 feet of the N-Reactor. The railroad is, however, DOE controlled and used in support of the operations at Hanford reservation. Since all other adjacent DOE production facilities have been shutdown, it basically serves the N-Reactor.
Tsunami	3	The N-Reactor site is inland and therefore this event is not applicable
Toxic Gas	4	This event is included under transportation accidents, on-site chemical release, and industry and military facilities accidents
Turbine Generated Missiles	2	The commercial power production related turbine generators are located in Building 185-N, about 650 feet from the 105-N and 109-N buildings. The rotation plane for these turbine blades is oriented parallel to 105-N and 109-N buildings so that the trajectory of fragments does not intersect these buildings. The possible turbine missile hazard from the onsite power generators is similarly eliminated. For such favorably oriented turbine generators, based on historical rates of failure, the probability of turbine missile strike will be less than 10^{-7} /year (Ref. 4). Moreover, the confinement system structure consists mainly of massive reinforced concrete

Table 3.1

Initial Screening of External Events for N-Reactor
(Continued)

Event	Applicable Screening Criteria*	Remarks
Volcanic Activity	1	<p>walls, floors and roofs and is not vulnerable to damage from turbine missiles. Other important safety related components have also been shown to be safe from turbine missiles (Ref. 2)</p> <p>All new category I structures, systems and components are evaluated for a design basis ashfall as given in Reference 2. However, ashfall loading does not govern the plant design.</p> <p>N-Reactor experienced a small (3/16 in) volcanic ashfall caused by the May 18, 1980 Mt. St. Helens eruption. Local emergency plant procedures have since been developed to protect the plant from repetition of this type external event. These procedures require immediate SCRAM upon notification of an eruption. Additional intake air filtration equipment for protection of personnel and plant equipment has been designed and is available on site. Oil bath air cleaners have been installed on the ECCS diesels and other local engine driven equipment. Procedures include allowance for change-out of dry filter media located downstream from the oil bath filters for these systems while continuing operation. Bonneville Power Administration transmission lines and local power distribution networks continued to operate during the 1980 event in other areas such as Ritzville and Yakima. Ashfall accumulations of as much as six inches were experienced at these locations</p>
Waves	4	This event is considered under external flooding

Table 3.1

Initial Screening of External Events for N-Reactor
(Concluded)

<u>Event</u>	<u>Applicable Screening Criteria*</u>	<u>Remarks</u>
--------------	---	----------------

* Notes

1. The event is of equal or lesser damage potential than the events for which the plant has been designed
2. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events
3. The event cannot occur close enough to the plant to affect it
4. The event is included in the definition of other event
5. The event is slow in developing and there is sufficient time to eliminate the source of the threat or to provide an adequate response

3.4 Extreme Winds and Tornadoes

3.4.1 Introduction

In this section, an approximate bounding analysis is applied to the risk from extreme winds and tornadoes. The probabilistic models used in such a bounding analysis integrate both the randomness and uncertainty associated with loads, responses, and capacities to predict the annual frequency of plant damage using conservative models. If the mean frequency derived from a conservative model is predicted to be low (e.g., $<10^{-6}$ /year), the external event may be eliminated from further consideration.

Often, some simplifications to the above analysis are introduced. As an example, back-face scabbing of the exterior barrier of safety related structures in case of tornado missile impact can be assumed to lead to reactor damage even though a sequence of failures is necessary to lead to this damage state. For some external events, it is possible to perform a bounding analysis without a structural response analysis. In effect, one could show that the frequency of exceeding design loads is very small. In addition to calculating a best estimate frequency of reactor damage, the uncertainties in hazard and component fragilities may be used to find the high confidence (95 percent) frequency of damage. An uncertainty analysis is required only if the external event leads to a best estimate damage frequency which is close to the (typically used) mean rejection frequency of 10^{-6} /year.

3.4.2 Bounding Analysis for Extreme Winds and Tornadoes

Extreme winds from tornadoes, hurricanes or windstorms can present a threat to nuclear reactors. The coast of Washington State is, however, not susceptible to hurricanes and in any case, N-reactor site is well inland and further protected from the influence of the oceanic winds by high mountain ranges. Hence, it is sufficient to consider risk to the structures at the site due to tornadoes and straight winds only. In general, tornadoes control the risk at higher wind speed but with a lower frequency, whereas straight winds from windstorms are considered to be the controlling wind level at a lower wind speed but with higher frequency. A typical wind/tornado hazard plot is shown in Figure 3.4 taken from Reference 5.

3.4.2.1 Tornadoes

A tornado is a violently rotating column of air, pendant from the base of a convective cloud, accompanied by damaging winds at the structure level. Tornadoes can be characterized by their rate of occurrence, direction, intensity, path length and path width. They are usually classified according to their intensity. The most common classification is based on the Fujita F scale and Pearson length and width scale (FPP) which provides a measure of the destructiveness of a tornado (Ref. 6). In this scale, tornadoes are assigned a number from 0 to 6 (F0-F6) with higher numbers indicating higher intensity tornadoes. Table 3.2 shows

PROBABILITY OF EXCEEDING
THRESHOLD WIND SPEED
IN ONE YEAR

Figure 3.4 Typical Tornado, Hurricane and Straight Wind Hazard Probability Models. (Note that relative positions of straight wind and hurricane models could be interchanged for different sites)

Table 3.2

Intensity, Length, Width and Area Scales

Scale No.	Fujita - F Intensity Scale (mph)	Pearson - P Length Scale (mi)	Pearson - P Width Scale (mi)	Area Scale (mi ²)
F-0	72	1.00	0.010	0.001
F-1	73-112	1.00-3.15	0.010-0.031	0.001-0.009
F-2	113-157	3.16-9.99	0.032-0.099	0.010-0.099
F-3	158-206	10.0-31.5	0.100-0.315	0.100-0.999
F-4	207-260	31.6-99.9	0.316-0.999	1.000-9.999
F-5	261-318	100-315	1.00-3.15	10.00-99.99
F-6	319-380	316-999	3.16-9.99	100.0-999.9

the classification of tornadoes along with intensity scale, length scale, and width scale. Also listed in Table 3.2 is an area intensity scale which is based on total damage area.

The damage from a tornado may be due to pressure differentials between the inside and the outside of a tornado, the missiles generated by a tornado or direct damage due to dynamic wind loadings. The F-scale intensities are assigned using a qualitative assessment of the worst damage that occurs during a tornado. This is usually accomplished by observing the damage to residential buildings or other structures and calculating the pressure that is needed to cause the observed damage. From calculated tornado wind pressure, one can find the maximum velocity which could generate such a pressure and this leads to the assignment of F-scale.

N-reactor is located in southeastern Washington state on the south bank of the Columbia River. The site is located in Region III (Figure 3.5) of the USNRC tornado risk regionalization scheme given in Markee et al. (Ref. 7). It is also located in Region D of the four region scheme proposed by Twisdale and Dunn (Ref. 8) as shown in Figure 3.6. According to both of these classifications, the region has the lowest tornado occurrence rate in the continental United States. For example, region III has a regional occurrence rate (corrected for unreported tornadoes) of $1.35\text{E-}05$ /square mile/year which is comparable to the rate of $3.53\text{E-}05$ /square mile/year for region D.

Ramsdell and Andrews (Ref. 9) give (Figure 3.7) the number of tornadoes occurring in the United States by 1° box (i.e., an area enclosed by one

Figure 3.5 Tornado Risk Regionalization Scheme Proposed by WASH-1300, Markee et al. (1975)

Figure 3.6 Tornado Risk Regionalization Scheme Proposed by Twisdale and Dunn (1983)

Hanford Reservation

Figure 3.7 Number of Tornadoes in U.S. by 1° Box (1954-1983)

degree latitude - longitude lines) from 1954 to 1983 as recorded by the National Severe Storms Forecast Center. Assuming the latitude for the site of 48° , the site specific frequency of occurrence in the local region will be around $1.0\text{E-}05/\text{square mile/year}$. According to Reference 2, the State of Washington experiences on an average, just one tornado annually, and no damage has ever been observed within 50 miles of the site. According to another data base (Ref. 10), the rate of occurrence of tornadoes within 100 miles of the site is 0.24/year, if all tornadoes are considered and 0.09/year if only the damage producing tornadoes are considered. This gives estimates of the rates of occurrence of $7.63\text{E-}05/\text{square mile/year}$ and $2.86\text{E-}05/\text{square mile/year}$, respectively.

Site specific studies for occurrence of tornadoes have been conducted by Dr. Theodore Fujita, a well known tornado expert (Ref. 11). The results of his studies indicate that tornadoes in the northwestern tri-state area (Washington, Oregon and Idaho) occur in distinctively specific alleys. Recent historical tornadoes in the three states are listed in Table 3.3 and plotted in Figure 3.8. It is seen that none of the tornadoes exceeded the F-3 level. Dr. Fujita re-evaluated his initial study in 1976 (Ref. 12) with essentially similar conclusions. In fact, all recorded tornadoes since 1950 have occurred in the alleys marked in Figure 3.8. The rate of occurrence of tornadoes even in these alleys is low.

The tornado alley nearest to the site is Walla Walla alley where only 6 tornadoes of F-2 or F-3 have occurred in 26 years. From Reference 11, the probability of maximum speed of 175 mph in Walla Walla alley is around $1.0\text{E-}7/\text{year}$ (Figure 3.9) and should be much lower than this at the plant site, since it is well outside the alley in which these tornadoes have occurred. Therefore, if the plant design is safe for up to 175 mph winds, the risk due to tornado hazard can be screened on the basis of low probability of occurrence. According to Reference 2, the plant has been evaluated for a design speed of 175 mph and all systems required for shutdown were found to survive this windspeed. Therefore, the risk to the plant from tornadoes is found to be negligible, subject to observations given under "plant design basis" in Section 3.4.2.3.

3.4.2.2 Straight Winds

The Hanford Meteorology Station, located on a plateau 280 ft above the Columbia River, 7-1/2 mile south of the site (elevation 433 ft) is the main source of wind data for the site. There was also an onsite meteorology tower, which has since been dismantled. Data from these two locations have been found to be consistent.

The dominant influence on the wind patterns at the site are the high Cascade mountains rising from 2,500 to 3,500 feet above sea level. These mountains serve as a source of cold air drainage causing a large diurnal range of speed during the summer. The actual wind conditions at this location depends on the topography, distance and direction from the ocean, and the position and intensity of the high and low pressure centers over the Pacific.

Table 3.3

List of Tornadoes in Three Northwestern States

<u>No.</u>	<u>Year</u>	<u>State</u>	<u>FPP</u>	<u>Length (mi)</u>	<u>Width (mi)</u>	<u>Area (mi²)</u>
1	1951	WA	321	8.0	0.02	0.16
2		OR	103	0.3	0.3	0.09
3	1953	WA	012	1.0	0.08	0.08
4	1954	ID	102	0.25	0.04	0.01
5		WA	102	0.5	0.06	0.03
6		WA	201	0.11	0.01	0.001
7		OR	101	0.5	0.01	0.005
8	1955	OR	113	2.5	0.2	0.5
9		ID	101	0.05	0.02	0.001
10	1956	WA	212	3.0	0.05	0.15
11		ID	101	0.1	0.03	0.003
12		ID	100	0.1	0.01	0.001
13		ID	100	0.1	0.01	0.001
14		ID	100	0.1	0.01	0.001
15		ID	200	0.1	0.01	0.001
16	1957	OR	?11	2.5	0.03	0.08
17		OR	233	18.0	0.15	2.7
18		WA	000	0.1	0.01	0.001
19		OR	000	0.1	0.01	0.001
20		ID	100	0.1	0.01	0.01
21	1958	OR	121	5.0	0.02	0.1
22		WA	000	0.1	0.01	0.001
23		OR	100	0.1	0.01	0.001
24		WA	201	0.24	0.03	0.007
25		ID	223	5.0	0.23	1.15
26		ID	202	0.2	0.05	0.01
27	1959	ID	102	0.4	0.06	0.024
28		ID	100	0.1	0.01	0.001

Table 3.3

List of Tornadoes in Three Northwestern States
(Continued)

<u>No.</u>	<u>Year</u>	<u>State</u>	<u>FPP</u>	<u>Length (mi)</u>	<u>Width (mi)</u>	<u>Area (mi²)</u>
29	1960	OR	211	1.0	0.03	0.03
30		ID	000	0.1	0.01	0.001
31	1960	ID	221	5.0	0.03	0.15
32		ID	000	0.1	0.01	0.001
33	1962	ID	231	10.0	0.01	0.1
34		OR	101	0.5	0.01	0.005
35		WA	000	0.1	0.01	0.001
36		ID	100	0.1	0.01	0.001
37		ID	000	0.1	0.01	0.001
38		WA	211	3.0	0.02	0.06
39	1964	ID	203	0.5	0.1	0.05
40		OR	000	0.01	0.01	0.001
41		WA	000	0.1	0.01	0.001
42		ID	000	0.1	0.01	0.001
43		ID	201	0.5	0.02	0.01
44		WA	121	8.0	0.02	0.16
45		OR-WA	111	1.0	0.01	0.01
46		WA	201	0.5	0.02	0.01
47	1966	WA	201	0.1	0.01	0.001
48		OR	201	0.25	0.03	0.003
49		OR	332	12.0	0.06	0.72
50		OR	101	0.1	0.01	0.001
51		WA	000	0.1	0.01	0.001
52		ID	221	5.0	0.01	0.05
53		ID	000	0.1	0.01	0.001
54		ID	000	0.1	0.01	0.001
55		ID	000	0.1	0.01	0.001
56		ID	000	0.1	0.01	0.001
57	1966	ID	000	0.1	0.01	0.001

Table 3.3

List of Tornadoes in Three Northwestern States
(Concluded)

<u>No.</u>	<u>Year</u>	<u>State</u>	<u>FPP</u>	<u>Length (mi)</u>	<u>Width (mi)</u>	<u>Area (mi²)</u>
58		OR	202	0.5	0.04	0.02
59	1968	ID	200	0.1	0.01	0.001
60		OR	234	10.0	0.6	6.
61		ID	201	0.5	0.025	0.012
62	1969	ID	100	0.1	0.01	0.001
63		WA	322	5.0	0.06	0.3
64	1970	WA	101	0.1	0.01	0.001
65		WA	101	0.25	0.02	0.005
66	1971	OR	101	0.4	0.01	0.004
67		WA	221	5.0	0.01	0.05
68		WA	122	3.5	0.05	0.18
69		WA	002	0.5	0.05	0.025
70	1972	OR	100	0.5	0.03	0.015
71		WA	323	9.0	0.3	2.7
72		WA	222	4.0	0.05	0.2
73		WA	000	0.1	0.01	0.001
74		WA	212	1.0	0.05	0.05
75		WA	212	1.0	0.05	0.05
76		ID	115	2.5	1.0	2.5
77		ID	111	1.0	0.01	0.01
78	1973	ID	100	0.5	0.1	0.05
79		ID	001	0.03	0.01	0.001
80		OR	003	0.5	0.1	0.05
81		OR	000	0.1	0.01	0.001
82	1974	OR	112	1.0	0.05	0.05
83	1975	OR	101	0.1	0.01	0.001
84		OR	000	0.1	0.01	0.001
85		OR	214	2.0	0.1	0.2

Figure 3.8 Distribution of Tornadoes in Northwestern States (1950-1975)

Figure 3.9 Tornado Probability in N-Reactor Vicinity

The Hanford site can also experience high wind speeds due to squall lines, frontal passages, strong pressure gradients and thunderstorms. Topographic channeling results in prevailing WNW or NW winds most of the time. However, winds from the southwest are the strongest. April is the month with the highest average speed of 11.1 mph. Table 3.4 gives the 50 foot level monthly average and peak gust speeds from 1945 to 1982. Return periods for the wind are given in Figure 3.10.

The highest average speed ever recorded was 80 mph in 1972. The annual probability of non-exceedance for a 175 mph wind velocity (assuming a Type I extreme value distribution with parameters computed from Figure 3.10) is substantially lower than the mean frequency of $1.0E-6$ /year which is considered to be the screening level used in this bounding analysis for external events. Therefore, the risk to the plant due to straight winds is negligible.

Table 3.4

Monthly Average Speeds and Peak Gusts at HMS
During 1945-1972 Recorded at 50-ft Level

Month	Average Speed MPH	Highest Average MPH	Lowest Average MPH	Peak Gust Speed MPH
JAN	6.4	10.3	3.1	80
FEB	7.0	9.4	4.6	63
MAR	8.4	10.7	5.9	70
APR	9.0	11.1	7.4	60
MAY	8.8	10.5	5.8	71
JUNE	9.2	10.7	7.7	72
JULY	8.6	9.6	6.8	55
AUG	8.0	9.1	6.0	66
SEPT	7.5	9.2	5.4	65
OCT	6.7	9.1	4.4	63
NOV	6.2	7.9	2.9	64
DEC	6.0	8.3	3.9	71

Figure 3.10 Peak Wind Gust Return Periods

3.4.2.3 Plant Design Basis

A site with a low frequency of extreme winds may still have a high risk if all plant structures and other outdoor equipment are not built to the same standards or some other factors increase the likelihood of damage and compromise safety. This is identified by a review of plant design bases and a plant walkdown.

The N-reactor started operation in 1963 and the original design assumed a maximum straight wind load of 78 mph (Ref. 2) at a height of 50 feet above ground level. The structures were designed using the Uniform Building Code of 1961 with some additional conservatism. All the design, construction, inspection and testing were performed in accordance with applicable codes of the time. There have been new developments in all these areas and it is necessary to evaluate the plant design for the new design criteria. The design passed one such test when the structures and systems located within them survived intact a peak recorded gust velocity of 80 mph in January, 1972. However, a review of the plant design indicated that, until the plant standards were revised in 1974 to include a design basis tornado (DBT) with a maximum wind speed of 175 mph., the plant was not designed for tornado loading. The DBT specified was as follows:

- a. Maximum horizontal speed - 175 mph
- b. Rotational speed - 150 mph
- c. Translational speed - 25 mph
- d. Pressure drop-in 3 seconds - 0.75 psi

In addition, the evaluation of the effects of tornado-generated missiles for a spectrum of standard missiles was required. Based on the investigation of the surrounding region, the missiles specified that could result from a 175 mph tornado were as follows:

- a. 2 in. x 12 in. wood plank, 12 ft. long, at 100 mph
- b. 4 ft x 8 ft. plywood sheet, 3/4 in. thick, at 150 mph
- c. 26 in. x 20 ft. sheet of 20 gauge corrugated steel siding, at 150 mph

Automobiles, railroad flatcars and similar heavy compact objects were assumed not to become airborne at this DBT. In this scoping study, it is assumed that there will not be an early warning of a tornado approach and both the offsite and on-site power supply systems will not survive the direct impact of a tornado on the N-reactor. Both of these are realistic assumptions because of the nature of tornado onset and its likely force. In such a case, reactor cooling will transfer automatically to the ECCS.

The critical facilities required for safe shutdown of the reactor and the operation of the ECCS are shown in Figure 3.3. These include 105-N (the building housing the main control room), 182-N (high-lift pump house), 181-N (river pump house and gantry crane), 116-N (ventilation stack), 109-N (heat exchanger building), and other auxiliary areas like 163-N, 183-N and 184-N. The central part of 105-N plus a fraction of area 109-N is designated the reactor confinement area. The confinement envelope surrounds the reactor coolant system and helps in maintaining its structural integrity. At a confinement pressure of 10 inches of water, the fog spray stations in 105-N and pipe gallery are automatically actuated. The anticipated behavior of these areas and systems important for the operation of ECCS and the effect of tornado wind pressure and missiles on these is described next.

3.4.2.3.1 Buildings 105-N and 109-N

The reactor with its control and trip systems is located in 105-N. In addition, this building contains auxiliary equipment, gas dryers, a water storage basin for spent fuel pool, electric switchgear, motors, fans, compressors, pumps, valves and piping. Building 109-N contains the reactor cooling pumps, the steam generators, the pressurizer (located in a penthouse), graphite shield cooling components and other auxiliary equipment. The common confinement structure is made of reinforced concrete walls varying from three to five feet in thickness and which provide shielding and pressure confinement capability. Since the walls and roofs were originally designed for an internal confinement pressure greater than the expected tornado uplift forces, the wind loads produced by the design basis tornado should not damage the concrete confinement structure. Similarly, missiles associated with a tornado are also unlikely to penetrate the thick walls of the confinement.

The confinement in 105-N is surrounded on three sides by other rooms and adjoins 109-N confinement on the south side. The control room is well inside the building complex, with walls at least 12 inches thick. These will not be subject to direct wind forces unless all of the outer area is damaged. Even then, these walls have been evaluated to survive the wind forces produced by the DBT. Missiles are unlikely to reach such locations, and even if they do reach them, the probability of penetration and subsequent damage is minimal due to substantial obstructions being present which would prevent a direct impact.

Most of the outer walls of 105-N and 109-N buildings are constructed of an insulated-metal covering supported by steel frames. Roof trusses support roof elements. These structures were designed for wind speeds ranging from 78 mph to 95 mph. Direct impact of a DBT on these structures is likely to severely damage this area but still will not result in subsequent damage to the concrete confinement structure or the reactor components located therein.

3.4.2.3.2. Building 181-N (River Pump House)

The river pump house is a reinforced concrete structure which contains seven deep well turbine pumps. Three of these support the ECCS system and four are normally operating. The pumps, two large diesel engines and auxiliary equipment required for ECCS operations, are located on the operating floor enclosed in a structure with reinforced hollow concrete blocks.

A wind velocity of 95 mph was used in the original design, however, earthquake forces were found to control the design of the building. The rigid frame of the concrete superstructure (including the roof slab, beams, girders and the 12 columns) is designed to resist lateral loads independent of the 12 inch thick concrete filler reinforced block walls. The walls are anchored to the floor with dowels and to the beams at the top. It has been calculated (Ref. 13) that even if a tornado with wind velocity of 166 to 175 mph hits directly on these walls, they will not collapse. There are two hatch covers, each 81 square feet, located on the roof. It is assumed that these will lift during a tornado to equalize pressure differential associated with a 0.75-psi drop, thus not subjecting the walls to this differential pressure for which the pump house was not designed.

While the pump house building has adequate wind resistance and venting capability for tornado conditions, some equipment on the operating floor is vulnerable to tornado missile damage. The current N-Reactor Accelerated Safety Enhancement Program (ASEP) provides for limited tornado protection upgrades to the pump house within 24 months of plant re-start. Required upgrades include:

- a. Protection for the batteries, diesel fuel lines, power panel boards, diesel engines and pumps, and control circuits located within the 181-N Emergency Pump Room
- b. Protection for the exposed fuel oil lines where they enter the Emergency Pump Room

3.4.2.3.3. Building 182-N

This building contains the five diesel driven pumps serving the ECCS and the confinement fog-spray, high pressure injection pumps, low pressure raw water and high pressure raw water pumps and associated piping. The basement walls, columns and ground floor slab are constructed of reinforced concrete. The walls above the ground floor, approximately 21 feet high, are made of concrete block, anchored to the structural steel superstructure. The block walls are reinforced with vertical steel bars and horizontal steel mesh located between each row of blocks. Precast roof planks are supported on steel beams. Even though the superstructure is capable of resisting up to 166 to 175 mph winds, the roof and the

block walls are likely to collapse at the DBT. Critical components of the ECCS are, however, located in the basement and will not be damaged by tornado winds or missiles. Similarly, the diesel fuel oil and day storage tanks which supply the ECCS low lift and high lift diesel driven pumps are also located underground and hence not vulnerable to tornado damage.

ASEP tornado missile protection upgrades are also required for certain ECCS equipment on the main floor of the 182-N building. Required upgrades include protection for the batteries, power panels, and conduit.

3.4.2.3.4. Ventilation Stack 116-N

The 116-N ventilation stack is a reinforced concrete stack, 200 feet high above the ground level, with an outside diameter of 15 feet at the top and 22.1 feet at the base. The stack is designed for a peak gust velocity of 104 mph at a height of 50 feet, 114 mph at 100 feet and 123 mph at 200 feet. Design studies (Ref. 13) show that a 175 mph tornado will overstress the stack and it is likely to fail. It may be possible for the broken parts to land on 117-N filter building. However, both 116-N and 117-N are not required for safe shutdown of the reactor. The broken parts from the stack are unlikely to damage 105-N, since it is located reasonably far to the south.

3.4.2.4 Summary

In summary, it is assumed that both offsite and onsite power will be lost during a design basis tornado, and the ECCS will be required. An evaluation of the ECCS for tornado wind pressures and missiles showed that the critical systems can survive up to 175 mph winds and the missiles generated (provided that the ASEP tornado missile protection upgrades are implemented). The site has some structures that are vulnerable to tornadoes; for example, most of the outer parts of Buildings 105-N and 109-N, ventilation stack 116-N and Building 182-N are likely to be extensively damaged from the direct impact of a tornado. But this damage would not prevent safe shutdown of the reactor.

3.5 External Flooding

External flooding is concerned with risk to the plant from all sources of water that are located outside the plant. In general, regulatory guidance on the topic for nuclear power plants is given in 10 CFR 50, Appendix A and 10 CFR 100 supplemented by Regulatory Guides 1.27, 1.59 and 1.102. Evaluation guidelines for DOE facilities are given in Kennedy et al. (Ref. 14).

In the licensing process, the methodology for safety assurance from inundation due to external flooding involves estimation of a parameter

called the design basis flooding level (DBFL) and comparing it with the elevation of critical safety related components and structures necessary for safe shutdown. The DBFL is generally the maximum water elevation attained by the controlling flood including coincident wind effects. The computations are site specific, as discussed in the regulatory documents cited earlier.

At the N-reactor site, the potential sources of flooding are:

- a. Flooding due to severe local precipitation and runoff (on the site itself),
- b. Flooding in the Columbia river due to precipitation runoff, ice jams, etc., and
- c. Flooding due to dam failure from overtopping, ice jam, seismic forces or some random cause.

In addition, appropriate combinations of these have to be considered in any flooding analysis.

The aspects of flooding which need to be considered are the frequency of the flood initiating event and the response of the plant. In the case where the frequency of the initiating event itself is sufficiently small, further study of plant response is not necessary. Otherwise, a systems analysis of the plant is required to assess the plant response. Two alternative approaches available for the systems analysis are a bounding analysis or a fully developed PRA methodology. Only the bounding type approach will be used here.

3.5.1 Plant Site Flooding From Severe Local Precipitation

The local site drainage area (Figure 3.11) is 800 acres. It has a 6,000 feet river frontage and extends about one mile landwards. The drainage area is limited by topographic features that include Hanford Ditch providing natural drainage to the east and precluding drainage from the south entering the site. The site has an elevation of 453 feet, which is 70 feet above the river bed.

Rainfall at the site is 6.28 inches annually, with three months in winter contributing 43 percent of the total. Maximum daily precipitation of one inch or more has occurred occasionally. The average return period for various precipitation amounts and intensities based on extreme value analysis of 23 years of data is given in Table 3.5. This table shows the concentration of rainfall in a given time-interval. For example, for a 100 year return period, the rainfall varies from 0.81 inch in one hour, 1.07 inches in three hours, 1.59 inches in six hours to 1.99 inches in a day. Compared with these, the highest intensity storm on record produced 0.59 inch in one hour, 1.08 inches in three hours, 1.68 inches in six hours and 1.88 inches in 12 hours. Approximately 39 percent of the

Table 3.5

Average Return Periods for Various Precipitation Amounts and Intensities

Return Period (Years)	Amount (Inches)					Intensity (Inches Per Hours)						
	Time Period (Hours)					Time Period (Hours)						
	1	2	3	6	12	24	1	2	3	6	12	24
2	0.26	0.30	0.36	0.48	0.62	0.72	0.26	0.15	0.12	0.08	0.052	0.030
5	0.40	0.48	0.55	0.77	0.95	1.06	0.40	0.24	0.18	0.13	0.079	0.044
10	0.50	0.59	0.67	0.96	1.17	1.28	0.50	0.30	0.22	0.16	0.098	0.053
25	0.62	0.74	0.83	1.21	1.45	1.56	0.62	0.37	0.28	0.20	0.121	0.065
50	0.72	0.85	0.96	1.40	1.66	1.77	0.72	0.42	0.32	0.23	0.138	0.074
100	0.81	0.96	1.07	1.59	1.87	1.99	0.81	0.48	0.36	0.27	0.156	0.083
250	0.93	1.11	1.22	1.82	2.13	2.26	0.93	0.55	0.41	0.30	0.177	0.094
500	1.02	1.22	1.33	2.00	2.34	2.47	1.02	0.61	0.44	0.33	0.195	0.103
1000	1.11	1.33	1.45	2.20	2.55	2.68	1.11	0.67	0.48	0.37	0.212	0.112
Existing Record	0.59	0.88	1.08	1.68	1.88	1.91	0.59	0.44	0.36	0.28	0.157	0.080

precipitation in winter is in the form of snow, with as much as six inches of snow accumulation on the ground. The probability of exceedance of a given snow depth at the site is given in Figure 3.12.

The intensity of local precipitation at the site can be described in terms of the probable maximum precipitation (PMP) which consists of intense heavy precipitation over a relatively short period of time. It is usually associated with small areas or drainage basins since such intense precipitation cannot be sustained uniformly over a larger watershed. The PMP evaluation procedure has been developed by the U.S. Weather Bureau and reported in Hydrometeorological Reports No. 33 and 43 (Refs. 15, 16). The method used in NRC Regulatory Guide 1.59 also relies on these documents. The design PMP for the site was determined to be up to 11.6 inches in six hours. This is far beyond any value actually observed or likely for even a 1,000 year return period (Table 3.5) and no flooding of this area has ever occurred. Due to the limited drainage area and excellent drainage paths to the river, flooding due to local precipitation can be eliminated from further consideration.

Entrances to the buildings are located at 454.5 feet, i.e., 1.5 feet above ground level and this is considered adequate. Building 182-N may possibly be subject to local runoff but the flood drains and a sump provided in the basement area drain to the river by a 42 inch pipeline. The building wall penetrations are sealed and moreover, these are either located several feet above or below ground. Therefore, it is concluded that the area covered by 100-N is not subject to flood damage from either rain or fast melting snow since adequate drainage paths are provided to carry the discharge to the river.

3.5.2 Flooding Due to Precipitation Runoff

N-reactor is located on the Columbia river approximately at river mile 380. The river is the predominant hydrological feature of the area with a clearly marked bed. The Columbia river drains a total area of 258,000 square miles in Canada and the United States. It has been extensively developed with some 72 dams on the river and its tributaries upstream of the site. The flood flow in the river at the site may thus be due to precipitation runoff from the basins or discharge from some of the reservoirs behind the dams. This section will consider the runoff from precipitation and dam regulation. Floods due to possible dam failures are considered in the next section.

The flow in the river varies depending on the season and amount released from the upstream dams. The highest levels of flow mostly occur in May or June due to a combination of rapid snowmelt and rain. The regulated low winter flows range from 36,000 cfs to 160,000 cfs. The high flows, typically occurring in spring, vary from 250,000 cfs to 450,000 cfs. The largest flood recorded near the site was in 1948 with a peak discharge of 690,000 cfs. The largest historical flood (1894), based on high water marks, is estimated to have been around 800,000 cfs.

Figure 3.12 Greatest Depth of Snow on Ground (24 Year Date)

As the river becomes more regulated, the flood potential is liable to change. Over the years, a number of reports have considered the potential for flooding at the site (e.g., Ref. 17). The estimates in all the previous reports are based on consideration of probable maximum flood (PMF) and upstream dam failure. McCann and Boissonnade (Ref. 18) have performed a probable maximum flood assessment in accordance with recent Department of Energy Guidelines (Ref. 14) and recent safety review recommendations by the N-Reactor Review Panel (Ref. 19).

The PMF is an extreme flood that represents the flood discharge that may be expected from the most severe combination of meteorological and hydrological conditions that are possible in the region. The Corps of Engineers has conducted many PMF studies over the years for the site by methods outlined in its 1969 report (Ref. 20). It estimates the unregulated PMF at the site to be 1,600,000 cfs and a regulated PMF flow of 1,440,000 cfs. Construction of other dams on the river and its tributaries provides additional storage to reduce the flow and hence the unregulated PMF value may be considered to be the conservative upper bound at the site from hydrological considerations, i.e., excluding dam failures.

A statistical study of the flood discharge data at river mile 396 (Table 3.6), just upstream from the site was conducted (Ref. 18) using Weibull, Type I External, Two Parameter Lognormal, Three Parameter Lognormal, Pearson Type III and Log Pearson Type III distributions. The resulting hazard curves are shown in Figure 3.13. Based on these curves, the probability of exceedance of the upper bound discharge of 1,600,000 cfs will range from $3E-10$ to $2E-6$ per year.

Besides uncertainty in the choice of distribution to fit the observed data, there is also considerable uncertainty in the parameters of these distributions computed from this data, due to the limited period of record. In Reference 18, these uncertainties in the model and its parameters were aggregated by a suitable weighting scheme to produce a final family of five discrete curves with equal weights (Figure 3.14). The mean probability of exceedance of 1,600,000 cfs was found to be $5E-7$ per year.

Thus, the flood hazard due to hydrological events can be screened out on the basis of low frequency of occurrence. In addition, it must be reiterated that these floods are dependent on extensive rain in the catchment area over a period of four to eight weeks, and can be forecast accurately by Columbia River Flood Control System. As this type of flood event is slow in developing, adequate warning is available to shutdown the reactor.

3.5.3 Flooding in the River Due to Dam Failure

The Columbia river and its tributaries have been extensively regulated by dams and reservoirs, both in Canada and the United States, to meet the

Table 3.6

Columbia River Gage Discharge Data at RM 396

<u>Year</u>	<u>Discharge</u> <u>(cfs) x 10³</u>	<u>Year</u>	<u>Discharge</u> <u>(cfs) x 10³</u>
1894	740	1951	408
1917	432	1952	369
1918	432	1953	395
1919	368	1954	420
1920	359	1955	448
1921	484	1956	553
1922	424	1957	427
1923	429	1958	397
1924	339	1959	438
1925	434	1960	325
1926	197	1961	547
1927	471	1962	319
1928	523	1963	316
1929	347	1964	472
1930	270	1965	370
1931	259	1966	335
1932	387	1967	506
1933	508	1968	377
1934	404	1969	314
1935	378	1970	279
1936	413	1971	360
1937	290	1972	431
1938	393	1973	266
1939	298	1974	384
1940	274	1975	246
1941	173	1976	282
1942	321	1977	192
1943	355	1978	203
1944	239	1979	195
1945	366	1980	275
1946	425	1981	316
1947	368	1982	322
1948	692	1983	307
1949	406	1984	254
1950	525	1985	214

Figure 3.13 Mean Frequency-Discharge Curves

Figure 3.14 Flood Discharge Hazard Curves

requirements of flood control, hydroelectric power generation, irrigation, municipal and industrial water supply, etc. On the main stream of the river in the United States, there are seven dams upstream and four dams downstream of the site as given in Table 3.7. It can be seen that Priest Rapids Dam is the nearest dam influencing flow at the site.

There are, in all, 72 dams of various sizes upstream of the site, the important ones being listed in Table 3.8. Not all of the 72 dams upstream of the site present any hazard to the N-Reactor site. This is because many of these are small dams and their failure would only have a local impact. Based on the dam break analysis studies performed by Corps of Engineers and US Bureau of Reclamation (Refs. 21, 22, 23) as part of emergency management planning or dam safety evaluations, only eight out of 72 dams are considered (Ref. 18) likely to cause flooding at the site. In order of their increasing distance from the N-Reactor site, these eight dams are: Priest Rapids, Wanapum, Rock Island, Rocky Reach, Wells, Chief Joseph, Grand Coulee and Mica. The data characterizing these dams is given in Table 3.8. Among these, Grand Coulee is the most upstream U.S. dam (Table 3.7). Only Mica in Canada is further north; 638 miles from the site.

Potential failure of any of these upstream dams could result in extreme flows, which might cause cascading failure of other dams. Therefore, study of dam failures upstream of the site is an important aspect of flooding analysis at the N-Reactor site. There is no generally accepted methodology for this purpose (Ref. 5). Ideally, such a study would be based on detailed examination and the analysis of the individual dams. Due to the complexity of such a task for a large interconnected system, site specific bounding calculations based on generic data, historical rates of failure, engineering judgment, etc., were performed for different modes of dam failure.

Dams may fail in a variety of modes which may be classified either as external or internal. External causes include dynamic, short term loads that are applied to dams due to earthquakes, aircraft crash, meteorite impact, overtopping, wind-wave action, sabotage, etc. Internal failure events are those due to static, operational loading conditions which might cause static structural failures, foundation failure, piping, embankment stability failure, etc., and these events collectively are referred to as random dam failure modes. Dam failure due to both seismic and random events is considered in this study since other external failure modes are of negligible frequency.

3.5.3.1 Random Dam Failure

No realistic engineering evaluation methods are available to predict the rates for random dam failures. Therefore, historic rates of failure are computed from the catalog of dam failures (Ref. 24). As the data base is

Table 3 7

Dams On the Columbia River with Potential
to Cause Flooding at N Reactor

	<u>Dam</u>	<u>River Mile</u>	<u>Approximate River Miles From Site</u>
Upstream:	Priest Rapids	397	17
	Wanapum	416	36
	Rock Island	453	73
	Rocky Reach	474	94
	Wells	516	136
	Chief Joseph	545	165
	Grand Coulee	598	218
	Mica	1018	638

not robust, it is not possible to evaluate the failure rates for specific types of dams. Over the years, a number of studies have been concerned with floods at the N-Reactor site due to dam break. A recent USBR study summarizes the previous work and estimates flood flows between 2,000,000 to 8,000,000 cfs at the site due to a break in the Grand Coulee Dam.

In a dam failure probabilistic assessment by McCann and Boissonnade (Ref. 18), event trees were formulated for each dam to consider the combination of failure events that might lead to flooding at the site, incorporating uncertainties in the reservoir level, spillway gate operation, break size and the likelihood of downstream dam failure due to overtopping. Floods at downstream locations were estimated for each failure scenario by using National Weather Service computer program NWS DAMBRK, assuming geometry, reservoir characteristics, roughness coefficient, spillway design, response of other downstream dams, etc. The failure rates for individual events were updated with subjective assessment by using Bayesian analysis applied to historic data.

Table 3.8

Major Upstream Dams

River	Dam	River Mile	Type of Structure	Height Above Stream Bed (ft)	Active Reservoir Capacity 1000 ac.-ft	Total Storage 1000 ac.-ft
Columbia	Priest Rapids	397	Concrete Gravity and Earthfill	100	44	250
Columbia	Wanapum	416	Concrete Gravity and Earthfill	133	161	500
Columbia	Rock Island	453	Concrete Gravity	73	9.5	100
Columbia	Rocky Reach	474	Concrete Gravity and Earthfill	140	36	300
Columbia	Wells	516	Concrete Gravity and Earthfill	72	74	250
Columbia	Chief Joseph	545	Concrete Gravity	205	116	450
Columbia	Grand Coulee	598	Concrete Gravity	355	5,200	9,500
Pendoreille	Albeni Falls	90	Concrete Gravity	66	1,153	---
Southfork Flathead	Hungry Horse	5	Concrete Arch	520	3,160	---
Clark Fork	Kerr Dam	72	Concrete Arch	186	1,219	---
In Canada	Arrow	781	Concrete Gravity and Earthfill	170	7,090	7,100
In Canada	Mica	1,081	Rockfill	640	12,000	20,000
Duncan	Duncan	8.3	Earthfill	130	1,400	1,400
Kootenai	Libby	220	Concrete Gravity	370	5,000	6,000

The occurrence of floods at a site due to a given combination of parameters is itself random. Therefore, the net result of event tree analysis is a flood hazard curve for each failure sequence in the event tree. The flood hazard at a given flood level for the failure of each dam is equal to the sum of the hazard at that level for each sequence (Figure 3.15). Similarly, the flood hazard considering the random failure of all dams on the Columbia river is derived from the hazard curves for each dam.

A review of Figure 3.15 shows that the random flood hazard is dominated by a failure of two dams (Mica and Grand Coulee) and that the most extreme flood discharges occur if either of these two dams fail. The result of a sensitivity study indicates that the dominance of the failure of one of these two dams in the hazard of the N-reactor site is so preponderant that other aspects of the analysis have no impact on the frequency of exceedance of flooding at the plant.

3.5.3.2 Seismic Induced Dam Failure

All of the U.S. dams in the N-reactor region have been designed for 0.1 g seismic acceleration based on the dams being in Zone II of Seismic Risk Map of the Uniform Building Code. Canadian dams are designed for 0.15 g. Grand Coulee, being the largest U.S. dam, has undergone considerable scrutiny regarding its seismic safety. An Energy Research and Development Study (Ref. 25) concluded that seismic damage to Grand Coulee will not produce a flow rate in excess of the PMF at the N-Reactor site. The U.S. Bureau of Reclamation conducted seismic analysis with the design basis earthquake of 0.25 g horizontal and 0.17 g vertical developed for the Fast-Flux Test Facility (FFTF) and essentially reached a similar conclusion.

A study of regional seismicity by John Blume Consulting Engineers (Ref. 26) conservatively estimated the maximum horizontal ground motion of 0.08 g for the operating basis earthquake and 0.16 g for the safe shutdown earthquake. Both these values are less than the design basis earthquake for which the Grand Coulee dam has been analyzed.

Other consulting engineers' reports (Refs. 27, 28, 29) have analyzed the seismic risk to the Chief Joseph, Priest Rapids and Wanapum dams and in all cases, the conclusion is that the dams have adequate safety factors for the various seismic conditions investigated. Table 3.9 gives the seismic design capacity of dams and the minimum factor of safety obtained by conservative analysis methods. In all cases, the seismic capacity values are higher than earthquakes likely to occur at the site.

A probabilistic evaluation by McCann et al. (Ref. 18) using generic assumptions, dam fragilities and a seismic hazard curve (Figure 3.16), estimated the seismic failure rates for the dams (Table 3.10). As many dams in the system may be subjected to the same earthquake, the possibility of multiple failures was investigated. The probabilistic study assumed that only three pairs of dams can fail jointly:

Figure 3.15 Flood Hazard Curves for Random Failure of Dams

Table 3.9

Seismic Design Capacity

Dam	Seismic Load Peak Ground Acceleration (g)	Type of Analysis	Minimum Factor of Safety
Mica	0.15 (H)	Finite element static analysis and dynamic response analysis, based on H.B. Seed's shear slice method, using the 1952 - Taft, California, earthquake record scaled to the maximum credible earthquake's peak horizontal acceleration.	1.3
Grand Coulee	0.23 (H) 0.09 (V)	Finite element static and dynamic response analysis using the 1933 Vernon, California, earthquake scaled to a magnitude 7.5 maximum credible earthquake at 50 miles and response spectrum.	2.0
Chief Joseph	0.25 (H) 0.17 (V)	Finite element static and dynamic response analysis using response spectrum.	1.5
Wells	0.20 (H)	Static and dynamic response analysis.	1.2
Rocky Reach	≥ 0.10 (H)	N/A	1.2*
Rock Island	≥ 0.10 (H)	N/A	1.2*
Wanapum	0.10 (H)	Pseudo-static finite slope method.	1.2*
Priest Rapids	0.10 (H)	Pseudo-static infinite slope method.	1.2*
Estimated

Figure 3.16 Seismic Hazard Estimate for the Region in the Vicinity of Dams on the Columbia River

Table 3.10

Seismic Failure Rates

Dam	Frequency (Per Year)
Mica	5.43-5
Grand Coulee	6.88-6
Chief Joseph	1.33-5
Wells	5.75-5
Rocky Reach	7.16-4
Rock Island	7.16-4
Wanapum	7.16-4
Priest Rapids	7.16-4

(1) Chief Joseph - Wells, (2) Wanapum - Priest Rapids, and (3) Rocky Reach - Rock Island. No other multiple failures were considered likely.

Using these single and multiple dam failure rates, the scoping analysis performed by McCann, et. al., showed that certain dam failures could result in water levels at the N-Reactor site which could pose a risk to the plant (Figure 3.17). A vital area analysis was conducted on all N-Reactor buildings to determine what accident sequences would arise as a result of flooding. The level of all the structures is given in Table 3.11. All but one of the safety related structures are flood protected at least up to 454 feet - 6 inches with 453 feet as the site elevation. The sole exception is the intake structure (181-N) which has an operating deck at 421 feet. The low lift diesel driven pumps are located in this structure with two in the main building and one in the annex. The two pumps located in the main building are 5 feet above the operating floor. Located immediately outside the intake structure are four CRW pumps, two on either side of the building.

The vital area analysis revealed that a flood exceeding the operating level of the intake structure would fail all three frontline cooling systems and thus result in a (GAC) transient sequence. Failure of the low lift diesel pumps will fail the ECCS (A) system. Failure of the CRW pumps can affect both the PCS/SCS (G) and the GSCS (C), since the CRW pumps provide secondary side heat exchanger cooling for the recirculation mode of the GSCS. In addition, these same pumps provide suction for either the high pressure raw water or fog spray diesel pumps and thus fail the once-through mode of the GSCS.

Since flooding of the intake structure along with the CRW pumps can fail all core cooling systems and their elevation is 28 feet below that of any other building, no further analysis of any other structures was required. In the probabilistic study of McCann (Ref. 18), the frequency of exceeding the operating level of the intake structure was estimated to be $7.0E-4$ per year. However, upon close analysis of the individual dam hazard curves in Reference 18, the total frequency of exceeding the operating level of the intake structure was found to be $3.9E-4$ per year with Wanapum Dam contributing $2.4E-4$ per year and Grand Coulee and Mica Dams together contributing $1.5E-4$ per year.

After a careful review of the emergency response guideline for failure of upstream dams it was determined that no procedural provisions have been made for an alternate source of water to be available as a heat sink for the three frontline cooling systems in the event of flooding of the 181-N Intake Structure and loss of the CRW and low lift diesel pumps. Therefore, initially no credit for recovery of the plant from this external flooding scenario could be given.

However, after discussions with the N-Reactor engineering staff, a procedure is planned for completion prior to plant startup to allow for alternate sources of cooling water from either the B or D area storage

Figure 3.17 Flood Hazard Curves for Seismically Induced Dam Failure

Table 3.11

N-Reactor Site Elevations of Important Structures and Equipment

Location	Elevation ft-msl
Plant grade	453
River pump house floor	421
Low-life diesels and pumps	426
Reactor control room floor	454
Emergency cooling and auxiliary pump house	454.5
Water storage tanks	454.5
Reactor and steam system buildings	454.5
Fuel oil storage tank	460
Diesel oil storage tanks	455
Electrical supply equipment	454.5

basins. These water sources will remove the single area flooding vulnerability of the 181-N Intake structure and the CRW pumps. A cross connection is already physically in place in the basement of building 182-N between the export water line and the suction header for the fog spray diesel pumps.

A sample procedure has been analyzed in detail and applied directly as a recovery action to the modified core damage frequency estimates from Reference 18. Flood arrival time estimates are based on References 18 and 20. A conservative velocity of 13 mph was used to calculate an arrival time of 2.7 hours from Wanapum (located 3 miles from N-Reactor) and 16.7 hours from Grand Coulee (located 218 miles from N-Reactor). A further degree of conservatism in the recovery analysis allowed the same flood arrival time from Mica as Grand Coulee even though it should be approximately twice as long.

With non-recovery probabilities in 2 hours (for Wanapum) and in 16 hours (for Grand Coulee and Mica) applied to frequency of exceedance estimates for the intake structure the total core damage frequency was found to be $4.7\text{E-}5$ per year.

The McCann study (Ref. 18) also estimated the frequency of flooding exceeding plant grade was $7.0\text{E-}5$ per year. In this case a transient GAC sequence was found to occur. However, no recovery from this scenario can be allowed.

In general, the risk result quoted above is conservative. However, the degree of conservatism cannot be quantified until further information is obtained. A careful review of Reference 18 along with personal communications with the author identified a number of specific areas where significant conservatisms are thought to exist. Any one of these areas discussed below could result in significant reduction in the overall flooding risk at the N-Reactor site.

Potential conservatisms include the following:

- a. In assessing seismic hazard curves for use in estimating seismic dam failure frequencies, data provided by Canadian authorities for the Mica Dam and by the U.S. Geological Survey were used. Given the time frame when these hazard curves were estimated, it is strongly suspected (and author McCann agrees) that the earthquake catalogs utilized included all earthquakes above a Modified Mercalli (MM) Intensity of 4.0. More recent work sponsored by the U.S. Nuclear Regulatory Commission (Ref. 30) has shown that earthquakes below MM 5.5 are not damaging to engineered civil structures. Hence, the early seismic hazard estimates used for both Mica and Grand Coulee are probably substantially conservative. For example, the failure probability for the Mica Dam (the dominant contributor) given in Reference 18 is $5.8\text{E-}5$ per year using the early hazard curve. If this hazard curve is scaled (approximately) to include only earthquakes greater than MM 5.5

Chapter 4) then the Mica Dam failure probability is reduced to $1.8E-5$ per year. Thus, given that the Mica Dam is an important contributor to the seismically induced flooding risk, this would result in a factor of three reduction in overall risk.

- b. The conservatism in the design failure level of the Mica and Grand Coulee Dams analyzed in Reference 18 is unknown. Factors of 1.3 and 2.0 were applied to the design level of each of these dams to obtain an estimate of the actual dam failure level. Furthermore, the result obtained was assumed to correspond to a minus 1.0 log standard deviation. Thus, the actual assumed conservatisms for these two dams are 1.8 for Mica and 2.8 for Grand Coulee. In general, given the nature of the conservative practices used in dam design, it is likely that these factors of safety are still significantly conservative relative to the gross breach failure modes considered in Reference 18. To reduce this level of conservatism it will be necessary to obtain and review the original design documents for these two dams.
- c. The consequences of dam failure in terms of predicted flood levels at the N-Reactor site were estimated in Reference 18 using a model of the Columbia River watershed (obtained from the US Army Corps of Engineers) and a hydrologic computer code which predicted the downstream water level given dam breach size and upstream pond characteristics. This code considers only a symmetric stream channel. However, in the vicinity of the N-Reactor site itself there are significant elevation differences between the two banks, and a significant potential for any flooding to occur in the wide space on the river opposite the N-Reactor site. Such wide spread flooding will significantly reduce the water levels which can occur on the N-Reactor side of the river, as well as significantly change (delay) the time of maximum flooding. A similar situation exists approximately nine miles upstream of the site, and may exist at numerous sites further upstream. The net effect of this is that the maximum water levels predicted by the symmetric channel computer code are undoubtedly significantly conservative. This conservatism could be reduced by employing a computer model which more accurately models asymmetric river channels and flooding basins.

Given the above conservatisms, it is probable that the actual probability of seismically-induced dam failures causing flooding at the N-Reactor site is significantly less than the value quoted above.

3.6 Internal Flooding

3.6.1 Introduction

A nuclear power plant contains many potential sources of internal flooding and flood locations. In order to make the analysis of these

floods tractable, a process was defined and applied to N-Reactor to identify candidate sources and critical flooding areas and to estimate their contribution to core damage frequency if required. The process consisted of the following steps:

- Step 1 Identify the important flood sources and critical flooding areas during the initial plant walkdown. Critical areas can be thought of as those plant areas where flooding could not only result in a plant trip but also damage safety-related equipment needed to mitigate the effects on any potentially induced plant transient.
- Step 2 Identify all initiating events which have the potential to be induced by flood for each flood source in each critical area. This step of the analysis results in identifying the spectrum of potential flood rates and is also used in quantification of flooding initiating event frequencies.
- Step 3 Perform a screening analysis comprised of the following steps:
 - a. Eliminate all plant areas not identified either by the initial plant walkdown or by computer mapping of critical equipment.
 - b. Perform a computer-aided vital area analysis which allows for the incorporation of random failures (i.e., failures not related to the flood itself) as well as all flood related damage. This is a similar process to what occurs in the fire analysis (see Section 5.5) which should be referred to for more details on this procedure. This step resulted in identifying the flood zone singles and singles with random combinations listed in Table 3.12.
 - c. Screen out flood zones based on physical location of vital equipment within the flood zone or other plant-specific aspects. Flood zones can be screened from further analyses based on:
 - (i) The equipment vulnerabilities are power or control cabling which is located at least 8 feet above the floor elevation and flooding would preferentially occur at lower elevations.
 - (ii) No flooding sources are located internal to the flood zone of interest.
 - (iii) Flooding would traverse through either open grating or down stairwells to lower levels within the same structure.

Table 3.12 indicates which of the aforementioned considerations eliminated the flood zone from further detailed analysis. An

example from Table 3.12 is Zone 3B1 which is a corridor immediately outside Room 6. Only cabling which is at least 8 feet from the floor is located in this zone. Also, no flooding sources are located in this zone. Therefore, this zone was eliminated from further consideration.

d. Screen on frequency for each remaining flood scenario.

- (i) Quantify core damage sequences for each remaining flood scenario.
- (ii) Perform an uncertainty analysis utilizing the TEMAC computer code for all remaining scenarios.

Step 3 eliminated all areas from consideration except for the 182-N basement which is at elevation 430 feet. As in Section 3.5.4, flooding will flow into the sump that is 7 feet below the basement floor. The floor drains in the basement discharge directly to the river. Thus, as was the case for the probable maximum flood scenario, for internal flooding, this area can also be eliminated from further consideration. Hence, internal flooding is found to contribute negligible risk at the N-Reactor.

Table 3.12

N-Reactor Flooding Vital Area Analysis Summary

<u>Single Zones</u>	<u>Single Zone Plus Randoms</u>	<u>Double Zones</u>
Zone 3A (ii)	Zone 1D (i)	None
Zone 3B1 (i,ii)	Zone 3B (i,ii)	
	Zone 22 (i)	
Zone 4A (ii)	Zone 27A (ii,iii)	
	27B (ii,iii)	
Zone 21 (i,ii)	Zone 31B (iii)	
Zone 31A	Zone 31C (ii,iii)	
	Zone 32A (i)	

*Zones are identical to the fire zones defined in Table 5.3

3.7 Conclusions

A scoping quantification study was performed to determine the external events which should be included in the detailed PRA study. The scoping quantification study considered all possible external events at the site except for seismic and fire events, since these two events were included in a detailed external events analysis. The PRA Procedures Guide (Ref. 1) was used as a guideline for identification of all possible external events at the N-Reactor site. Next, an initial screening process was carried out to eliminate some of the events from the list. For this purpose, a set of screening criteria was developed and then each external event was examined for possible elimination based on these criteria. After the initial screening process was completed, the following events were found to be potential contributors to the plant risk:

- a. Extreme winds and tornadoes
- b. External flooding
- c. Internal flooding

A bounding analysis was performed for these events. The degree of sophistication in the bounding analysis for each event depended on whether the event could be eliminated based on only a hazard analysis or a complete analysis including hazard analysis, fragility evaluation and plant response analysis.

Tornadoes, extra tropical winds and tornado missile impacts were eliminated on the basis of low tornado strike probability and features of plant structures and components designed to withstand the effects of the Design Basis Tornado.

Internal flooding was eliminated based on a set of screening criteria outlined in Section 3.6.

External flooding due to local precipitation or hydrological events was found to contribute negligibly to the plant risk. Flooding due to random or seismically induced dam failure was found to be a potential risk contributor. This conclusion is, however, based on a number of generic assumptions, which may not reflect the actual condition of dams in the system.

3.8 References

1. U.S. Nuclear Regulatory Commission, PRA Procedures Guide, NUREG/CR-300, January 1983.
2. N-Reactor Updated Safety Analysis Report (NUSAR), UNI-M-90, UNC Nuclear Industries, Inc., Richland, Washington.
3. K. A. Solomon et al., Estimation of the Hazards to a Nuclear Reactor From the Random Impact of Meteorites, UCLA - ENG - 7426, March, 1974.
4. J. A. Lambright and M. P. Bohn, Analysis of Core Damage Frequency: Peach Bottom Unit 2 External Events, NUREG/CR-4550, Vol. 3, October 1989.
5. C. Y. Kimura, and R. J. Budnitz, Evaluation of External Hazards to Nuclear Power Plants in the United States, NUREG/CR-5042, December, 1987.
6. T. Fujita, and A. D. Pearson, Results of FPP Classification of 1971 and 1972 Tornadoes, Paper presented at the Eighth Conference on Severe Local Storms, October, 1973.
7. E. J. Markee, J. G. Beckerley, and K. E. Sanders, Technical Basis for Interim Regional Tornado Criteria, WASH-1300, U.S. Government Printing Office, Washington, May, 1974.
8. L. A. Twisdale, and W. L. Dunn, Probabilistic Analysis of Tornado Wind Risks, Journal of the Structural Division, ASCE, February, 1983.
9. J. V. Ramsdell, and G. L. Andrews, Tornado Climatology of the Contiguous United States, Pacific Northwest Laboratory, Richland, Washington, NUREG/CR-4461, PNL-5697, March, 1986.
10. H. G. Daubek, Tornado History and a Discussion of the Tornado Warning System, Battelle Memorial Institute, Pacific Northwest Laboratories, Report to Jersey Nuclear Company, Richland, Washington, December, 1970.
11. T. Fujita, Estimate of Maximum Wind Speed of Tornadoes in Three Northwestern States, SMRP Research Paper No. 92, University of Chicago, December, 1970.
12. T. Fujita, Re-evaluation of the Design Basis Tornado at the HPFL Facility, Richland, Washington, December 7, 1976.
13. H. S. Davis, Behavior of Critical Structures and Equipment at 100-N Plant During a Postulated Tornado, UNI-666, May 1, 1978.

14. R. P. Kennedy, S. A. Short, J. R. McDonald, M. W. McCann Jr., and R. C. Murray, Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazard, UCRL - 1590, Lawrence Livermore National Laboratories, Prepared for: Department of Energy, 1987.
15. NOAA/NWS, 1956, Seasonal Variation in the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24, and 48 Hours, Hydrometeorological Report No. 33 (1956).
16. U. S. Weather Bureau, Hydrometeorological Report No. 43, Probable Maximum Precipitation, Northwest States.
17. U. S. Army Engineer Division, North Pacific, Corps of Engineers, Portland, Oregon, letter dated February 7, 1972 to Mr. M. J. Hroncich, Burns and Roe, Inc.
18. M. W. McCann Jr., and A. C. Boissonnade, Probabilistic Flood Hazard Assessment for the N-Reactor Hanford, Washington, UCRL - 21069, July, 1988.
19. Department of Energy, N-Reactor Review Panel, 1986.
20. U. S. Army Engineer Division, North Pacific Corps of Engineers, Portland, Oregon, Lower Columbia River Standard Project Flood and Probable Maximum Flood, September, 1969.
21. U. S. Army Engineer, District Seattle Corps of Engineers, Artificial Flood Considerations for Columbia River Dams, August, 1963.
22. U. S. Army Engineer Division, North Pacific Corps of Engineers, Portland, Oregon, Artificial Flood Considerations for Columbia River Below Chief Joseph Dam to Richland, Washington, January, 1968.
23. Corps of Engineers, Washington, D. C., Studies Pertaining to Artificial Flood Possibilities on the Columbia River at Hanford, Washington, December 10, 1951.
24. M. W. McCann Jr., and G. Hatem, Progress on the Development of a Library and Data Center on Dam Incidents in the United States, Department of Civil Engineering, Stanford University, Progress Report No. 2 to Federal Emergency Management Agency, Washington, D. C., 1985.
25. Energy Research and Development Administration, Richland Operations Office, Evaluation of Impact of Potential Flooding Criteria on the Hanford Project, 1976.
26. John A. Blume and Associates, Engineers, Seismic Exposure and Criteria for Seismic Analysis of Grand Coulee Dam, Columbia River, Washington, Washington Public Power Supply System, Richland, Washington, November 1971.

27. Cornell, Howland, Hayes and Merryfield Consultants, Priest Rapids Development - Stability Study of the Embankments and Reservoir Shoreline, Grand County Public Utility District, Ephrata, Washington, May, 1974.
28. Cornell, Howland, Hayes and Merryfield Consultants, Wanapum Development - Fourth FPC Safety Report, Grand County Public Utility District, Ephrata, Washington, June, 1975.
29. John A. Blume and Associates, Engineers, Predicted Flood Levels in the Richland Washington Area from Columbia and Yakima River Dam Failures, Exxon Nuclear Co., Inc., Richland, Washington, September, 1974.
30. R. P. Kennedy, Engineering Characterization of Ground Motion, NUREG/CR-3805, May 1984

4.0 SEISMIC PRA

4.1 Summary of Results and Conclusions

A detailed seismic risk assessment was performed for the N-Reactor plant. This analysis utilized dynamic response calculations for all important structures, a generic seismic fragility data base for components, and detailed component fragility derivations for a number of components identified during the plant visit as falling outside the generic data base. A hazard curve developed by the Woodward-Clyde Corp. was used. Point estimates and mean values of accident sequence and core damage frequencies were obtained using a Monte Carlo approach.

This section presents the results of the seismic risk analysis. Results for two cases are presented. The base case is our best estimate of the current configuration of the plant and its emergency procedures. In addition, results for a case involving a planned seismic upgrade of the 109-N building turbine bay (109-NT) are given. This building upgrade was identified and approved in the process of completing the N-Reactor seismic risk assessment. Building 109-NT as currently configured will fail at lower ground accelerations than building 182-N, thus tending to dominate the facility seismic risk profile. This building will be strengthened prior to any restart of the N-Reactor as described in the N-Reactor Seismic Task Force Analysis and Fix Summary (Ref. 1).

Seismic component failure probabilities were taken from a generic fragility data base and from site specific building and component fragility analyses. A total of six initiating events and three event trees were used for the seismic analysis as described later in this chapter.

A total of 13 accident sequences survived the conservative screening process. These 13 sequences are presented in Section 4.6 along with identification of the boolean sequences that were solved for each accident sequence. These 13 non-negligible accident sequences resulting from the screening were fully requantified using best-estimate random failure mean probabilities and best estimate seismic fragilities and responses plus associated variabilities. Based on this final quantification, five dominant sequences were identified. These dominant sequences are:

<u>Accident Sequence</u>	<u>Base Case</u>	<u>Upgrade 109NT Case</u>
BLDGS 182 AND 109-NT	63%	11%
Process Tube Rupture (PTR)	17%	34%
LOSP T8	12%	43%
SLOCA 9	4%	2%
SLOCA 8	3%	7%

as shown in Figure 4.1. For definitions of these sequences, refer to Section 4.1.1. The percentage contributions were taken from Monte Carlo

Base Case

(CDF = 4.6E-5/yr)

Upgrade 109NT Case

(CDF = 2.2E-5/yr)

Figure 4.1 Principal Contributors to Seismic Core Damage Frequency

uncertainty results. The total mean core damage frequency for the base case was computed to be $4.6\text{E-}5$ per year. The range factor (defined as the ratio of the 95th percentile to the 50th percentile of the distribution) on the base case total core damage frequency was found to be 39. For the upgraded Building 109-NT case, the mean core damage frequency is reduced to $2.2\text{E-}5$ and the uncertainty bands are slightly increased.

4.1.1 Description of Accident Sequences

To understand the dominant accident sequences, it will be recalled that there are essentially three mitigating systems identified on the event trees. System G consists of the primary and secondary coolant systems (PCS/SCS). The designator G denotes successful functioning in the first two hours whereas G1 denotes successful functioning of the system in the 2-24 hour time frame. Similarly, system A is the emergency core cooling system (ECCS) operating successfully in the 0-2 hour time frame. Designator A1 is the same ECCS system in a 2-24 hour time frame. The designator A2 denotes failure of one check valve on the inlet side of the core resulting in failure to cool 1/16 of the core. Finally, system C denotes the graphite shield cooling system (GSCS). The GSCS system is assumed to be effective in removing sufficient decay heat to prevent core damage only beginning two hours after shutdown (as was assumed in the internal events analysis). Whether or not the GSCS system fails within the first two hours after shutdown only determines the release category for the backend calculations. That is, core damage is assumed to occur whether or not the GSCS system works or fails. Finally, it will be recalled from Chapter 2 that for large LOCA situations no credit is taken for the primary and secondary cooling systems.

The dominant base case accident sequence results from the collapse of either buildings 182-N or 109-NT. This leads directly to accident sequence GAC. No other failures are required to cause this transient accident sequence. The PCS/SCS fails due to numerous support pumps for that system which are located in building 182-N and are assumed to fail given building failure. Similarly, the ECCS system fails because the high-lift diesel pumps are located in building 182-N. The GSCS fails because its make-up water pumps are located in building 182-N. Collapse of building 109-NT is presumed to destroy all control wiring for the GAC systems all of which is routed through the basement of this building.

The second most important base case sequence is the process tube rupture (PTR) sequence. This occurs due to failure of the reactor thermal shield wall at peak ground acceleration (PGA) levels above one SSE. Failure of the thermal shield wall leads to direct shearing of many process tube inlet and outlet pipes. This produces an uncoolable core configuration. No other failures are required to cause this accident sequence. The next three sequences in order of importance are the loss of offsite power sequence LOSP T8 and the small loss of coolant sequences SLOCA 8 and SLOCA 9. The LOSP T8 and the SLOCA 8 sequences result from the same

logical combination of cut sets and component failures following the (different) initiating events. In each case, the sequence involves GA-C system failures. The dominant cut sets contributing to these accident sequences consist of failure of the silo (which fails the ECCS system) in conjunction with electrical bus failures (both 13.8KV and 4KV) which fail the primary and secondary cooling systems. In these sequences, the GSCS system is not failed. Failure of the PCS/SCS is caused by a dependency on instrument air which in turn requires power from 4KV busses. These component failure combinations are far and away the dominant contributors to these three accident sequences.

4.1.2 Core Damage Frequency as a Function of Peak Ground Acceleration

Contributions of the dominant accident sequences to the seismic core damage frequency are dependent on the magnitude of the earthquake ground motion. Figure 4.2(a) shows the cumulative contribution of the dominant accident sequences (as a percent of the total core damage frequency) for the base case. As can be seen, the building 182/109-NT dominant accident sequence makes a significant contribution above 0.25g (the 1 SSE level) and dominates the total core damage frequency. The loss of offsite power sequence also contributes at these earthquake levels. The remaining dominant sequences make contributions above the 2 SSE level.

Figure 4.2(b) shows the same contributions assuming the Building 109-NT fixes have been completed. In this case, significant contributions of the dominant accident sequences occur do not occur until 0.35g and above, and the Building 182/109-NT failures are no longer dominant contributors. Thus, considerably more margin exists in this configuration as reflected in the reduced frequency of core damage.

4.1.3 Engineering Insights

The importance of each basic seismic failure event was evaluated by (sequentially) setting the seismic failure probability for that event to zero in the mean point estimate calculation. This provides a measure of the net reduction in core damage frequency that would occur if that component could never fail due to seismic shaking. Results of these calculations are shown in Table 4.1. It can be seen that, for the base case, the largest reduction occurs for failure of building 109-NT due to the fact that it fails PCS/SCS and ECCS directly. The second most important contributor is the silo. Failure of the silo results in loss of the water sources for the high lift diesel driven pumps which fails the ECCS directly. Other contributors include LOSP (which includes failure of the ceramic insulators in the yard as well as buildings 151, 153 and 183). These all have approximately the same seismic capacity and effectively fail offsite power and the A train of AC power. Finally, failure of the core inlet piping makes a contribution since it leads to the process tube rupture event directly. All other components and structures have risk reduction potentials of less than 1 percent.

Figure 4.2(a) Contributions of each PGA level to Total Core Damage Frequency - Base Case

Figure 4.2(b) Contributions of each PGA level to Total Core Damage Frequency - Bldg 109NT Upgrade Case

Table 4.1

Dominant Component Contributions to Mean Core Damage
Frequency Ranked by Risk Reduction Potential

<u>Component</u>	<u>Percent Reduction if not Failed</u>	
	<u>Base Case</u>	<u>Upgrade 109NT Case</u>
Building 109-NT	56%	6%
Concrete Silo	10%	32%
LOSP	5%	11%
Core Inlet Piping	3%	13%
Building 182-N	1%	2%
4 KV Buss	1%	1%
13.8 KV Buss	1%	1%
Shield Wall	1%	3%

4.1.4 Other Engineering Insights

The electrical "A" and "B" busses at N-Reactor are not considered "safety grade" systems because of the reactor's reliance on diesel driven ECCS pumps and emergency DC battery supplies. Switchgear and distribution transformers for both busses are currently unanchored. It is recommended that anchorage be provided for these components capable of withstanding ground motions of 3 SSE with a high level of confidence prior to plant restart. This would be relatively inexpensive to implement, and would consist primarily of installation of anchor bolts to resist overturning and stops to preclude tipping and sliding.

In addition, the existing inventory of components should be reviewed for plant specific vulnerabilities such as transformer core supports or other unusual equipment configurations. Results of the PRA discussed herein are based on the existing unanchored electrical bus equipment.

4.1.5 Comparison to Operating Commercial Reactors

The seismic risk assessment methodology used for the N-Reactor analysis is relatively new. This method has recently been applied to two other commercial nuclear power plants which were part of the NUREG 1150 program (Refs. 2,3). These plants are Surry Unit 1, a Westinghouse three-loop PWR which came on line in 1972, and Peach Bottom Unit 2, a General Electric BWR-4 reactor which came on line in 1974.

Figure 4.3 provides a comparison between the seismic risk at N-Reactor for the "base case" reported above and the Surry and Peach Bottom commercial power reactors. This figure illustrates that the N-Reactor seismic risk is comparable to that of two somewhat more recent commercial power reactors. The comparison is made to the seismic risk for these plants using Lawrence Livermore National Laboratory estimates (Ref. 4) for the seismic hazard at each plant location. Lower seismic risks have been reported for these plants using an Electric Power Research Institute estimate (Ref. 5) for seismic hazard at each site.

4.2 Introduction

The seismic risk assessment performed for the N-Reactor plant utilized methodology developed and applied in the NRC-sponsored NUREG-1150 studies. These studies were aimed at providing a state-of-the-art update of our perception of risk at existing commercial power plants, and thus, were based on the most recent data and methods available. As summarized in Chapter 1, the seismic risk assessment was performed in seven steps:

- a. Determine the local earthquake hazard (hazard curve and site spectra),
- b. Identify accident scenarios for the plant which lead to radioactive release (initiating events and event trees),
- c. Determine failure modes for the plant safety and support systems (fault trees),
- d. Determine the responses (accelerations or forces) of all structures and components (for each earthquake level),
- e. Determine fragilities (probabilistic failure criteria) for the important structures and components,

Figure 4.3 Comparison of N-Reactor Seismic Risk to NUREG 1150 Commercial Nuclear Power Plants. (Upper and lower box limits are 95th and 5th percentiles, respectively).

- f. Compute the probability of core damage using the information from Steps a. through e. and,
- g. Estimate uncertainty in the core damage frequencies.

Each of these steps are described in the following sections.

4.3 Seismicity and Hazard Curves

The earthquake hazard at a given power plant site is characterized by a hazard curve and a site ground motion spectra. The hazard curve is a frequency plot which gives the probability of exceedance (per year) of different peak ground accelerations. The site response spectra describes the relative frequency content of the earthquakes expected at the site, and also the influence of the local soil column and layering in modifying the earthquake frequencies transmitted to the site.

4.3.1 General Considerations

For a given site, the hazard curve is derived from a combination of recorded earthquake data, estimated earthquake magnitudes of known events for which no data are available, review of local geological investigations, and use of expert judgment from seismologists and geologists familiar with the region in question. The region around the site (say within 100 km) is divided into zones, each zone having an (assumed) uniform mean rate of earthquake occurrence. This mean occurrence rate is determined from the historical record, as is the distribution of earthquake magnitudes. Then, for the region under consideration, an attenuation law is determined which relates the ground acceleration at the site to the ground acceleration at the earthquake source, as a function of the earthquake magnitude. The uncertainty in the attenuation law is specified by the standard deviation of the data (from which the law was derived) about the mean attenuation curve. These four pieces of information (zonation, mean occurrence rate and magnitude distribution for each zone, and attenuation law) are then combined statistically to compute the hazard curve.

The low level of seismic activity and the lack of instrumental records make it difficult to carry out seismic hazard analyses using historic data alone. To augment the data base, current methodologies make use of the judgment of experts familiar with the area under consideration. Approaches used to generate the subjective input, to assure reliability by feedback loops and cross-checking, and to account for biases and modes of judgment are described in detail in Bernreuter (Ref. 6).

4.3.2 Hazard Curves Used For N-Reactor

The hazard curves used in the N-Reactor PRA was derived from a study performed by Woodward-Clyde Corp. for the N-Reactor site (Ref. 7). This

study provided three mean hazard curves as a function of instrumental peak ground acceleration, corresponding to three values of lower bound magnitude used in deriving the curves. These curves are shown in Figure 4.4. The data base of seismotectonic studies used as the basis for this study was essentially identical to that used for the Washington Public Power Supply System (WPPSS) Final Safety Analysis Report (Ref. 8).

Since there is general agreement among seismic design engineers that earthquakes of Modified Mercalli (MM) magnitude less than 5.5 are not damaging to engineered structures (Ref. 9), a single hazard curve corresponding to lower bound magnitude of 5.5 MM was constructed by interpolating between the 5 MM and 6 MM curves derived by Woodward-Clyde (shown in Figure 4.4).

The empirically derived 5 MM and 6MM hazard curve are very nearly linear on the semi-log plot as shown, so for computational purposes the derived 5.5 MM curve was fit to a log-linear equation in the form

$$\text{Log(Frequency of Exceedance)} = -3.30 - 3.99 \cdot \text{PGA}$$

where PGA is peak ground acceleration. The resulting hazard curve corresponding to 5.5 MM is also shown in Figure 4.4. This mean hazard curve (along with an estimate of its associated uncertainty) was used in all the risk calculations for the N-Reactor seismic risk assessment.

Since no uncertainty bounds were available in the Woodward Clyde study, an estimate of the uncertainty in the seismic hazard at the N-Reactor site was constructed by assuming that the uncertainty distribution of the hazard at each peak ground acceleration is log normal, and taking the uncertainty parameter of the log normal distribution (beta-u) from other studies. Reference 3 is a recent study of 65 reactor sites in the Central and Eastern United States from which reasonable estimates of hazard curve uncertainties can be inferred. A beta-u value of 2.0 was used at 0.20g and a beta-u value of 2.5 was used at 0.75g. The resulting distribution of hazard curves is shown in Figure 4.5 where the mean, the median, the 15th percentile and 85th percentiles are shown. Tables 4.2 and 4.3 list the numeric values for the mean and median hazard curves resulting from fitting the hazard curves in this fashion.

Note that the mean hazard curve of Figure 4.5 is, in fact, above the 85th percentile hazard curve. This is due to the skewness of the log normal distribution which describes the uncertainty at each PGA value, and is typical of the results found in other seismic hazard studies. However, regardless of the uncertainty distribution assumed, the mean hazard curve determines the mean core damage frequency estimates as explained in Section 4.6. Hence, as long as the mean hazard curve is properly fit, the mean core damage frequency estimates will be correct.

Figure 4.4 N Reactor Mean Hazard Curves

Figure 4.5 Comparison of Mean, Median, 15% and 85% Hazard Curves Used In This Study For N Reactor

Table 4.2
Mean Hazard Curve CCDF Values

<u>PGA(g)</u>	<u>Annual Frequency of Exceedance</u>
0.05	3.18E-4
0.15	1.27E-4
0.25	5.04E-5
0.35	2.00E-5
0.45	7.96E-6
0.55	3.17E-6
0.65	1.26E-6
0.75	5.01E-7

Table 4.3
Median Hazard Curve CCDF Values

<u>PGA(g)</u>	<u>Annual Frequency of Exceedance</u>
0.05	5.85E-5
0.15	1.90E-5
0.25	6.15E-6
0.35	1.99E-6
0.45	6.46E-7
0.55	2.09E-7
0.65	6.79E-8
0.75	2.20E-8

4.4 Response Calculations

4.4.1 Introduction

As described above, a seismic probabilistic risk assessment (PRA) can be considered in a series of steps: seismic hazard characterization, seismic response of structures and components, structure and component failure descriptions, plant logic models, and probabilistic failure calculations. Section 4.4 deals with the frequency characteristics of the free field ground motion (an element of the seismic hazard characterization), the seismic response of structures and components, and the fragility evaluation of N-Reactor structures.

In a seismic PRA of a nuclear plant's safety systems, only the components affecting the operation of the systems and those structures housing or supporting these components need to be analyzed. Plant logic models identify the components. Plant general arrangement and mechanical drawings are then used to locate the components and identify the relevant supporting structures. For the N-Reactor facility, specific safety related components are housed in the reactor building 105-N, the heat exchanger bays and pipe gallery of building 109-N, and its annex building 109-NT, the river pump house building 181-N and building 182-N. In addition, electrical distribution equipment for off-site power is located in buildings 151, 153 and 184. Refer back to Figure 3.3 for the site plan.

Building 105-N is the reactor building at the Hanford site. The structure houses the reactor as well as support equipment and peripheral facilities. The structure is divided into five areas; however, only Area 2 is of interest to this study. Area 2 is a rectangular configuration in plan, approximately 213 feet long (east-west direction) and 140 feet wide (north-south direction). The entire area is separated from the adjoining portions of the building by a continuous seismic gap, either 2 or 3 inches wide, between adjoining walls and commencing at the top of a common wall footing. The peripheral areas would have a second order effect on the reactor response and therefore were not included in the soil-structure interaction (SSI) analysis. Grade in this area of the site is at elevation 454 ft. The foundation system consists typically of individual spread footings under the principal structural walls, except for a large common footing pad under the main process area supporting the reactor base and portions of the walls adjacent thereto. The bottoms of the footings are located between elevation 432 (22 feet below grade) and elevation 424 ft. An integral part of the reactor building is the process area approximately 42 feet by 47 feet located near the center of the structure. The process area consists of the inlet/outlet shields, primary shield, reactor core, cooling tubes, and process tubes.

Building 109-N houses the reactor's pipe gallery, steam turbines and heat exchangers used in the production of steam. Building 109-N sits on the south side of the reactor building. The pipe gallery is in the north side of the building, the heat exchanger cells are in the center, and the steam turbine bay is on the south side. The steam turbine bay is separated from the rest of the structure by seismic joints and was not included in the structural model.

Building 109-N is 430 feet long in the east-west direction and 205 feet long in the north-south direction. The pipe gallery is 385 feet long and 35 feet wide running from the reactor building south wall to the shielding wall of the cell areas. The cells are 80 feet long in the north-south direction and 50 feet wide. The original structure was designed and constructed in the early 1960s with the cell 6 addition on the west end constructed several years later. Concrete walls separate each of the heat exchanger cells. These concrete walls are 5 feet thick and provide both bearing support for the roof, and shielding from radiation. The roof structure over the pipe gallery and cells is a concrete slab 3.5 feet thick over the cells and portions of the pipe gallery. At the center of the pipe gallery roof, concrete beams span the gallery and support a 3 foot thick concrete roof slab. Building 109-N is founded on strip footings at 22 feet below grade.

The buildings 105-N, 109-N, and the process area are coupled into a single structural model founded on a single rigid foundation for this study. The single rigid foundation assumption was evaluated in Reference 10 and found to be appropriate.

The 181-N river pumphouse houses components of the N-Reactor emergency core cooling system. Building 181-N consists of a substructure, a superstructure, and a gantry crane. The substructure includes all portions at and below the operating floor at elevation 421 ft, grade for this area of the site. It is of heavy reinforced concrete construction. The building is on a mat foundation that is 5 feet thick and 94 feet by 124 feet in plan. The top of the mat is at elevation 360 ft. The operating floor varies from 1 to 3 feet in thickness. Most of the major equipment components are mounted on the operating floor. Several shear walls ranging in thickness from 3 to 4 feet stand on the mat foundation and support the operating floor. Grade on the east side is located at elevation 421 ft. Backfill slopes down to the top of the mat foundation on the west side (elevation 355 ft).

The superstructure rises above the operating floor. It encloses the low-lift emergency core cooling system pumps and diesel engines. The superstructure is 37 feet by 51 feet in plan and about 18 feet high. The concrete roof slab varies in thickness between 6 and 8 inches, and has two 9 foot square hatch openings. The superstructure has twelve reinforced concrete columns with interconnecting beams. Reinforced concrete block (masonry) wall panels (12 inches thick) between the exterior columns enclose the superstructure.

The gantry crane travels in the E-W direction on rails mounted to the operating floor. The main hoist has a 60-ton capacity. The span between the tracks is 104 feet and the distance between end stops is about 120 ft.

Building 182-N is a roughly square composite structure building consisting of two main floors and a roof penthouse. The foundation is at elevation (-)24 feet 6 inches. This building contains a main equipment area, control room and switchgear area. Important equipment in this building includes the high lift diesel pumps for the ECCS, support pumps for the PCS/SCS, and the GSCS make-up water and once-through cooling pumps and auxiliary equipment. The basement level walls are reinforced concrete whereas the first floor walls are a composite of steel columns and reinforced concrete block walls.

Building 109-NT is an auxiliary structure connected to building 109-N. Like building 182-N, this is a composite structure involving steel columns and reinforced concrete block walls. This building contains the turbine drive motors for the PCS pumps, associated piping and instrumentation. In addition, control cables and wiring for the major safety systems (PCS/SCS and ECCS) run through this building.

Essential electrical equipment for offsite power is contained in buildings 151, 153 and 184. These are all low rise masonry wall (concrete block) structures. Failure of any one of these buildings would result in loss of offsite power. These buildings were included in the accident sequence models, but were not analyzed in detail as the more likely cause of loss of offsite power is failure of ceramic insulators in the switchgear or failure of the (unanchored) yard transformers. A conservative failure level for these buildings showed that the risk increment due to their failure was negligible given the current configuration of ceramic insulators and transformers at the site.

Seismic PRAs require as input best-estimate structural response, variations of response, and correlation of response. A seismic PRA considers earthquakes over the entire range of the seismic hazard curve; hence, seismic responses must be determined over this range. Often, seismic response determined as part of the plant design process is available. However, this data reflects the conservatism associated with the seismic design analysis methodology and considers only low seismic excitation levels.

To determine structural response at the higher excitation levels required by a seismic PRA, either the design analyses must be extrapolated or reanalyses of the structures must be made. For this study, analytical models of each structure, identified above as housing safety related components, were developed and used in a probabilistic

response analysis to determine the best-estimate seismic response of these structures. The structural models for the 105-N/109-N/process area and 181-N pumphouse used in this study are those from previous analyses documented in References 10 and 11, respectively. The response modeling and fragility evaluations for buildings 182-N and 109-NT were performed by Westinghouse-Hanford Co. as described in Reference 12.

The balance of Section 4.4 will describe and summarize:

- a. Site and seismic input characteristics
- b. Probabilistic response analysis
- c. Safety related component response
- d. Fragility of structures

4.4.2 Site and Seismic Input Characteristics

4.4.2.1 Site Description

The N-Reactor site is characterized as a deep soil site with layers of dense sand and gravel, and hard silt and clay. Finished grade exists at elevation 454 feet. The mean water table is at a depth of 83 feet (el. 371 feet) from the ground surface.

4.4.2.2 Soil Properties and Earthquake Definition

Two interrelated objectives for this portion of the investigation were to:

- a. Define strain compatible soil properties over the range of seismic excitation levels defined by the seismic hazard curves, and
- b. Define the seismic input motion for the probabilistic response analyses of the structures,

Three seismic excitation levels were explicitly considered as defined by their peak ground acceleration in the horizontal direction: 0.25g (1 SSE), 0.5g (2 SSE), 0.75g (3 SSE). They are denoted acceleration ranges 1, 2, and 3 in subsequent discussions. These excitation levels were treated explicitly--input motions and probabilistic response for other levels defined by the hazard curve can then be interpolated from the results.

In general, soil properties such as shear modulus and damping are a function of soil strain and consequently a function of excitation level, i.e., acceleration ranges one, two, and three defined above. With higher excitation levels, soil shear modulus tends to decrease while

soil damping tends to increase. Equivalent linear visco-elastic soil properties as a function of excitation level were developed using the program SHAKE (Ref. 13). The soil deposit is idealized as a series of horizontal layers. The soil profile and low strain soil properties for the N-Reactor site are given in Figure 4.6 and Table 4.4 as specified in Reference 14. The low strain soil properties characterize the site as moderately stiff with shear wave velocities varying from 850 fps at the surface to a maximum of 4400 fps at depths between 83 and 103 feet. The shear wave velocity of the layer below and above the 20 foot thick stiff layer are approximately 1/2 this value. The last layer below a depth of 218 feet is hard silty-clay with a low strain shear wave velocity of 3200 fps. Estimation of equivalent linear soil properties is preceded by defining the relationship between soil shear modulus and strain (shear modulus degradation curve), and soil material damping and strain. N-Reactor specific data is available, however, shear modulus reduction factors and material damping versus strain relations are given in Table 4.5 as recommended in Reference 15 and used in this study. Vertically propagating shear waves are assumed to be the wave propagation mechanism by which horizontal motion propagates to the soil free surface. Nominal strain compatible soil properties for the three acceleration ranges are given in Figure 4.7 and Table 4.6. The seismic input motion used to develop these values and to perform the probabilistic response analyses are discussed below.

The selected free-field motion for the probabilistic response analyses treated local site amplification indirectly for the N-Reactor site. Section 4.4.3 describes the probabilistic response analyses performed for each structural model. In each analysis, ten earthquake acceleration time histories were considered for each component direction of the 3-D models. The acceleration time histories are recorded motions of actual earthquakes from similar soil sites. The selected ten recorded earthquake motions (three components each) are listed in Table 4.7.

The N-Reactor site median acceleration response spectra (of the ten horizontal components for both directions) are shown in Figure 4.8. For comparison, the median response spectra from an ensemble of recorded motions at similar soil sites, soft to medium clay and deep cohesionless soil, as reported in Reference 16 are shown. The comparison between the N-Reactor site and the Reference 16 spectra shows that the frequency content and amplification for the median response (of each set of ten components) adequately represent expected motion at the N-Reactor site.

4.4.3 Probabilistic Building Response Analysis

In recognition of the uncertainty in seismic input, and soil and structure properties, and the objective to estimate median response conditional on earthquakes with a specified peak ground acceleration, probabilistic response analyses were performed on the N-Reactor

Figure 4.6 Generalized Soil Profile for the N-Reactor Site [14] (V_s = fps, ρ = lb/ft³)

Table 4.4

N-Reactor Site Low Strain Soil Properties Versus Depth (7)

Depth (feet)	(P) Wave Velocity (ft/sec)	(S) Wave Velocity (ft/sec)	Unit Weight (wet) (lbs/ft ³)	Poisson's Ratio	Young's Modulus (lbs/in ²)	Shear Modulus (lbs/in ²)	Bulk Modulus (lbs/in ²)
0-8	1,800	850	135	0.36	0.57x10 ⁵	0.21x10 ⁵	0.66x10 ⁵
8-28	2,000	1,000	135	0.33	0.78x10 ⁵	0.29x10 ⁵	0.78x10 ⁵
28-62	2,200	1,200	137	0.29	1.10x10 ⁵	0.43x10 ⁵	0.86x10 ⁵
62-83	6,600	2,850	140	0.38	6.80x10 ⁵	2.4x10 ⁵	9.90x10 ⁵
83-103	9,000	4,400	140	0.34	16.0x10 ⁵	5.8x10 ⁵	17.0x10 ⁵
103-158	6,600	2,600	125	0.41	5.1x10 ⁵	1.8x10 ⁵	9.3x10 ⁵
158-218	6,400	2,200	129	0.43	3.9x10 ⁵	1.4x10 ⁵	9.6x10 ⁵
218-250+	7,000	3,200	120	0.37	7.3x10 ⁵	2.6x10 ⁵	9.2x10 ⁵

Table 4.5

Proposed Soil Shear Modulus Versus Strain Degradation
and Damping Versus Strain Relationship for N-Reactor Site

Shear Strain (%)	$<1.00 \times 10^{-4}$	3.16×10^{-4}	1.00×10^{-3}	3.16×10^{-3}	1.00×10^{-2}	3.16×10^{-2}	1.00×10^{-1}	3.16×10^{-1}	1.00	3.16	>10.00
Clay: ($V_s > 2000$ fps)											
G/G _{max}	1.00	0.99	0.98	0.94	0.89	0.80	0.70	0.60	0.52	0.50	0.50
Damping (%)	1.25	1.25	1.50	2.50	3.50	5.00	7.50	11.0	15.0	15.0	15.0
Sand:											
G/G _{max}	1.000	0.984	0.934	0.826	0.656	0.443	0.246	0.115	0.050	0.050	0.050
Damping (%)	0.80	0.80	1.70	3.20	5.60	10.00	15.50	21.00	24.60	24.60	24.60

Note: G/G_{max} denotes the factors (at various strain levels) which have to be applied to the shear modulus at low shear strain amplitudes (here defined as 10^{-4} percent) to obtain the modulus at higher strain levels.

Shear Wave Velocity (ft/s)

Damping Ratio

Legend:
 Case 1-Low Strain:
 Strain Compatible:
 1 $\times$ SSE:
 2 $\times$ SSE:
 3 $\times$ SSE:

Figure 4.7 Variation of Soil Shear Wave Velocity and Damping Ratio With Depth

Table 4.6

Strain Compatible Soil Properties
for N-Reactor Site

Layer #	Thickness (ft)	Unit Weight k/ft ³	1xSSE		2xSSE		3xSSE	
			Shear Modulus (k/ft ²)	Damping	Shear Modulus (k/ft ²)	Damping	Shear Modulus (k/ft ²)	Damping
1	8	0.135	2,490	0.033	2,100	0.051	1,820	0.068
2	10	0.135	2,800	0.054	2,010	0.092	1,475	0.125
3	10	0.135	2,200	0.083	1,260	0.140	690	0.189
4	11	0.137	3,240	0.082	1,890	0.138	1,070	0.185
5	11	0.137	2,750	0.099	1,350	0.166	650	0.215
6	12	0.137	2,400	0.114	1,140	0.180	530	0.225
7	11	0.140	27,940	0.037	23,660	0.054	19,060	0.080
8	10	0.140	27,190	0.040	22,290	0.061	15,450	0.102
9	10	0.140	73,710	0.025	65,760	0.038	54,640	0.057
10	10	0.140	72,980	0.026	64,550	0.040	53,950	0.059
11	12	0.125	18,030	0.054	12,960	0.092	7,770	0.143
12	13	0.125	17,580	0.057	12,530	0.095	7,460	0.146
13	15	0.125	16,970	0.061	11,900	0.100	5,780	0.168
14	15	0.125	16,454	0.065	11,720	0.102	4,900	0.182
15	20	0.129	10,440	0.085	6,810	0.129	1,730	0.226
16	20	0.129	10,230	0.087	6,680	0.131	1,010	0.246
17	20	0.129	10,230	0.087	6,680	0.131	1,010	0.246

Table 4.7

Selected Free-Field Acceleration Time
Histories for Probabilistic Response Analysis

<u>Earthquake</u>	<u>Site</u>	<u>Date</u>
Western Washington	Highway Test Lab- Olympia	4/13/49
Eureka	Federal Building, Eureka	12/21/54
Eureka	Ferndale City Hall	12/21/54
Puget-Sound	Highway Test Lab- Olympia	4/29/65
San Fernando	8244 Orion Blvd Los Angeles	2/9/71
Northwest, CA	Ferndale City Hall	10/7/51
Kern County	Hollywood Storage Building Basement	7/21/52
Imperial Valley	Imperial Valley Irrigation District	5/18/40
Hollister	Hollister City Hall	4/8/61
Borrego Mountain	Imperial Valley Irrigation District	4/8/68

Figure 4.8

N-Reactor Site Median Free-Field of Ten Horizontal Components

structures housing safety related components. The methodology used is that of SMACS (Ref. 17) as implemented in the computer program CLASSI (Ref. 18) utilizing the substructure approach. The substructure approach to soil structure interaction is composed of the following elements: (a) specification of the free-field ground motion, (b) determination of the foundation input motion and impedances, (c) calculation of the dynamic characteristics of the structure, and (d) analysis of the coupled soil structure system. Each element as pertaining to the N-Reactor structural analyses is discussed below.

4.4.3.1 Free-Field Ground Motion

Specification of the free-field ground motion includes specifying its frequency characteristics, spatial variation, and control point. The frequency characteristics and spatial variation of the free-field motion were previously discussed above. The elevation at which the free-field motion is specified for each structure is called the control point. Generally, this would be the existing free surface at elevation 454 feet for the N-Reactor site for surface founded structures. The specific control point chosen for each structure is detailed below.

105-N/109-N/Process Area. Buildings 105-N/109-N/Process Area are modeled and analyzed as surface-founded structures founded at a depth of 22 ft. The control point is assumed to be at this location also. Thus idealization initially ignores the effect of the soil above this elevation. In-structure response is later reduced (based on previous studies) to account for this conservatism.

181-N Pumphouse. The control point location for the pumphouse is specified at the ground surface elevation of 454 ft.

Grade Mounted Equipment. The free field ground response spectra is used directly in the analysis of yard water storage tanks, etc.

4.4.3.2 Foundation Input Motion

The foundation input motion varies from the free-field motion for all cases except surface founded foundations (i.e., control point and foundation at the same elevation) subjected to vertically propagating shear and dilational waves. A scattering function relates the three translational free-field components to the six degrees of freedom on the foundation. The scattering function is frequency dependent and complex valued. For this investigation all waves are assumed to be vertically propagating. Variation between free-field motion and foundation input motion is due to the variation of free-field motion with depth and wave scattering at the soil-foundation interface of embedded foundations. The foundation is modeled as rigid and massless, thus points on the foundation are constrained to move according to its geometry in plan and depth of embedment.

105-N/109-N/Process Area. The buildings 105-N/109-N/Process Area are considered surface founded for the structural response analysis. Reduction of the in-structure response for the effect of embedment is discussed in Section 4.4.4.

181-N Pumphouse. The scattering functions were calculated previously using FLUSH (Ref. 19) which is documented in Reference 11.

182-N and 109-NT. The soil structure interaction calculations for these two structures were developed using the FLUSH soil structure interaction program. Both structures were treated as surface founded structures.

4.4.3.3 Foundation Impedances

Foundation impedances are the force-displacement characteristics of the soil. Foundation impedances depend on the soil layering and material behavior, frequency of excitation, geometry and embedment of the foundation. For a rigid foundation, the force displacement characteristics are uniquely defined by a 6 x 6 matrix, complexed valued and frequency dependent, relating a resultant set of forces and moments to the six rigid-body degrees of freedom of the foundation. The soil properties used to calculate the impedance functions are those shown in Figure 4.6 for 1 SSE. Impedance functions for the higher acceleration ranges 2 SSE and 3 SSE were derived by scaling 1 SSE impedance functions by the ratio of average shear wave velocity through the strain compatible soil column (Ref. 17).

105-N/109-N/Process Area. For the purpose of calculating impedances, the N-Reactor structures 105-N/109-N/process area are assumed to be surface founded and supported on a single rigid foundation.

The process area basemat is connected to building 105-N through two heavy shear walls which run east and west along either side of the basemat, thus causing both foundations to move as a single unit. Buildings 105-N and 109-N share a shear wall running east and west along their common boundary. The orientation of this wall would tend to cause the foundation of both buildings to move as a single unit in the east-west direction. However, the connection between the buildings in the north-south direction is not completely rigid.

A previous study presented in Reference 10 calculated the impedances for two separate foundations, coupled only through the soil, and the restraints caused by the connections between the building and foundations were included in the structural model.

Considering the foundations of buildings 105-N and 109-N as either separate foundations or a single rigid foundation yielded similar response. The stiffening effects of the structure essentially cause the structure/foundation system to behave in a rigid fashion. Therefore, a single rigid foundation was analyzed here.

181-N Pumphouse. The reinforced concrete mat foundation of the pumphouse is 124 ft x 94 ft in plan having a uniform thickness of 5 ft. With the substructure shear walls, it is embedded to a maximum depth of 66 ft. The equivalent embedment ratio is therefore about 1.1. It was concluded that structural response will be significantly affected by embedment. The procedure to include this effect was to first calculate foundation impedances for a rectangular foundation of the specified shape and dimensions using the computer program CLASSI (Ref. 18) and assuming it to be surface founded. These foundation impedances were corrected frequency-by-frequency for the presence of soil on the sides of the foundation. The procedure is explicitly defined in Reference 11, a previous pumphouse structural analysis study.

Buildings 182-N and 109-NT. The soil-structure interaction analysis for these two structures was performed using a coupled finite element soil and structure model directly (an alternative to the frequency dependent impedance approach) using the ANSYS code as described in Reference 12. Hence, no impedances had to be developed explicitly.

4.4.3.4 Structural Models Dynamic Characteristics

Structural dynamic characteristics are described by their fixed base eigen system and modal damping factors. Eigen systems, frequencies and mode shapes, are determined from fixed base finite element models. Beam elements model stiffness between floor levels. The contribution to lumped mass at each floor level is from the half height of the wall above and below, floor slab, and equipment at that floor. Nominal values of structure damping were in general taken to be 0.07, 0.085, and 0.10 (fractions of critical damping) for the three seismic acceleration ranges considered here. Block walls in the pumphouse were assigned damping factors of 0.10 for all three acceleration ranges. These were based on published damping values and assumed stress levels achieved.

105-N/109-N/Process Area. A coupled structural model of Building 105-N, Building 109-N, and Process Area was originally developed for use in the previous SSI analysis of Reference 10. Initially, a separate model was defined for each of these three major structures. The final coupled model was formed by joining the structures and the process area as dictated by structural details and load paths. A brief description of each structure and its respective model follows.

Building 105-N. The building construction is basically of reinforced concrete, except for steel roof trusses, the roof deck, some insulated exterior metal wall panels, masonry partitions, and miscellaneous interior structural steel platforms. The building is about 92 feet high extending from the base of the strip footing at a depth of 22 feet to the roof at a height of 70 feet above grade. The building contains partial floors at several levels around the main process area. The roof is supported by a system of 10-ft-deep steel trusses spanning in an east-west direction, supporting the roof deck on their top chords and a concrete floor on their bottom chords. The part of the building below the process area elevation contains the lower sections of the piping spaces, equipment rooms, pipe and ventilation tunnels, pits and access spaces, and the process area foundation slab.

Building 109-N. The portion of building 109-N contributing to the structural model is constructed entirely of reinforced concrete shear walls and floor slabs. The building is 109.5 feet high from the base of the strip footing to the top of the pressurizer enclosure. The main roof is 88.5 feet above the base of the foundation.

Process Area. The main process area lies within Building 105-N and consists of the inlet/outlet shield, the primary shield and the reactor core itself. The seismic model is essentially that documented in Reference 20.

105-N/109-N/Process Area Coupled Model. Structural elements connecting buildings 105-N and 109-N couple the response of the structures. This coupling is implemented by joining the two shear beam models with rigid links at appropriate elevations. Buildings 105-N and 109-N share a common wall that runs in the east-west direction. This wall is doweled along its height and therefore is fully capable of transmitting east-west shear. Although floor slabs do not exist at all elevations, the response interaction between the structures is best approximated by rigidly connecting the models in this direction from the footing at elevation -22 feet to the top of the wall at elevation 60 feet with respect to grade. For north-south response, floor and roof slabs couple the two structures at elevations -16 feet, 40 feet, and 60 feet. Rigid links connect the shear beam models at these elevations. The reactor core model is attached to the building 105-N model at elevation 0 feet, the top of the process area pedestal. The shear beam model of building 105-N used for these analyses is that from Reference 10. Details of the structural models are documented in Appendix C of Volume II of Reference 20.

Figure 4.9 shows the 3-D fixed base model used to calculate the nominal frequencies and mode shapes. The model's structural

Figure 4.9

N-Reactor 105-N/109-N/Process Area
Fixed Base Model

frequencies and percent mass participating in each mode are given in Table 4.8. Significant horizontal modes are 11.8 Hz N-S and 13.6 Hz E-W accounting for 71 and 61 percent of the mass respectively.

181-N Pumphouse. The 181-N Pumphouse structure, including the substructure, superstructure, and gantry crane, is represented here by the three dimensional analytical model documented in Reference 11. The substructure portion is 66 feet in height extending from the basemat to the operating floor. It is of reinforced concrete construction. The normal water level elevation is 390 feet or 35 feet above the base of the foundation. The effective water mass was taken as the rigid tank impulsive fluid mass. Fluid sloshing loads are small and were not included. The superstructure resides on the operating floor and is 18 feet in height. It has twelve reinforced concrete columns with inter-connected beams to form moment frames. Twelve-in.-thick reinforced concrete block wall panels between the exterior columns enclose the superstructure. These panels have vertical steel reinforcement and horizontal joint reinforcement. They are mortared into the concrete columns and doweled to the operating floor. Masonry bond beams at the tops of the panels are reinforced and doweled to the columns. The superstructure is represented by a refined finite element model to obtain member forces and panel floor response spectra directly from the dynamic analysis. Mass properties include the weight of the structure, roofing, and attached equipment. Elastic stiffness properties were assigned to all elements.

The gantry crane travels in the E-W direction on rails mounted to the operating floor. The main hoist has a 60-ton capacity. The span between the tracks is 104 feet and the distance between end stops is about 120 ft. To obtain response of the gantry crane, equivalent single degree of freedom models were attached to the operating floor in the dynamic model. Properties of the equivalent models were selected to match the fundamental mode responses of the gantry crane in the horizontal directions. Fundamental frequencies of 2.6 Hz and 3.7 Hz in the N-S and E-W directions were determined from a more detailed model of the gantry crane. Properties of this model were derived from a previous analysis of the gantry crane (Ref.21). The current gantry crane model contains refinement over the previous model to better represent its actual flexibility.

Figure 4.10 shows the 3-D fixed base model used to calculate the nominal frequencies and mode shapes of the pumphouse. Frequencies of vibration and percent mass participating in each mode are given in Table 4.9. Significant horizontal modes are 7.39 Hz in the N-S direction, accounting for 80 percent of the mass, and 18.48 Hz and 18.96 Hz in the E-W direction accounting for 31 percent and 46 percent of the mass respectively.

Table 4.8

105-N/109-N/Process Area Coupled Model Frequencies
and Percent Mass Participation

<u>Mode</u>	<u>Hz</u>	<u>E-W</u>	<u>N-S</u>	<u>Vertical</u>
1	4.75	1.1%	0.0%	0.0%
2	5.01	1.3	0.0	0.0
3	8.27	0.0	0.3	0.0
4	11.81	0.0	70.7	0.0
5	13.59	60.6	0.0	0.0
6	16.32	0.0	0.1	0.0
7	17.50	0.0	1.8	0.0
8	18.78	1.5	0.0	0.0
9	19.42	0.0	2.7	0.7
10	19.54	2.9	0.0	0.0
11	19.78	0.0	0.1	0.3
12	20.01	0.0	0.0	0.0
13	23.83	0.0	0.0	0.0
14	25.16	0.0	0.3	0.0
15	25.56	0.0	0.3	0.0
16	26.24	4.8	0.3	0.0
17	26.50	0.4	2.9	0.7
18	28.13	0.0	0.0	31.1
19	28.22	0.0	0.0	0.0
20	30.54	0.0	0.4	31.5
21	31.90	0.3	0.0	0.0
22	32.23	0.0	0.2	1.0
23	35.06	0.1	0.8	0.7
Total Mass Participation		73.0%	80.9%	66.0%

Figure 4.10 N-Reactor 181-N River Pumphouse
Fixed Base Model

Table 4.9

181-N Pumphouse Structure Model Frequencies
and Percent Mass Participation

<u>Mode</u>	<u>Freq (Hz)</u>	<u>E-W</u>	<u>N-S</u>	<u>Vert.</u>
1	2.63	0.000%	0.819%	0.000%
2	3.69	0.589	0.000	0.000
3	7.39	0.000	80.317	0.000
4	9.34	0.000	6.543	0.000
5	13.28	5.503	0.001	0.002
6	13.42	0.191	0.020	0.000
7	15.01	0.000	0.000	0.000
8	15.18	0.000	0.000	0.033
9	15.91	0.002	0.000	0.000
10	15.92	0.000	0.000	0.000
11	15.92	0.000	0.000	0.000
12	15.92	0.000	0.000	0.000
13	15.93	0.196	0.000	0.000
14	16.84	2.316	0.001	0.000
15	17.19	0.997	0.000	0.000
16	18.06	0.010	0.089	0.000
17	18.48	30.626	0.004	0.011
18	18.96	45.922	0.013	0.024
19	19.23	0.467	0.537	0.000
20	19.97	0.000	10.256	0.000
21	23.59	0.000	0.000	0.000
22	23.60	0.000	0.000	0.000
23	23.60	0.000	0.000	0.000
24	23.60	0.000	0.000	0.000
25	23.94	0.000	0.013	0.000
26	25.58	0.000	0.013	0.000
27	27.44	0.010	0.000	1.086
28	28.71	0.000	0.000	0.000
29	29.20	0.053	0.000	0.426
30	30.70	0.000	1.173	0.000
31	31.88	0.013	0.000	10.447
32	32.05	0.000	0.000	0.000
33	33.52	0.050	0.000	63.522
34	36.76	0.007	0.000	7.071
35	37.04	0.008	0.000	3.563
36	37.75	0.023	0.000	0.147
37	38.45	0.000	0.000	0.136
38	39.45	0.005	0.000	0.002
39	40.11	0.000	0.063	0.000
40	41.48	0.000	0.000	0.007
41	43.99	0.000	0.001	0.000
42	44.88	0.092	0.000	0.024

Table 4.9 (cont'd)

181-N Pumphouse Structure Model Frequencies
and Percent Mass Participation (Concluded)

<u>Mode</u>	<u>Freq (Hz)</u>	<u>E-W</u>	<u>N-S</u>	<u>Vert.</u>
43	45.02	0.006	0.001	0.004
44	45.15	0.003	0.001	0.005
45	45.44	0.007	0.000	0.830
46	46.23	0.049	0.000	0.067
47	46.30	0.000	0.000	0.000
48	46.41	0.042	0.000	0.217
49	46.93	0.053	0.000	0.639
50	48.27	11.121	0.000	0.186
Total Pct Mass		98.363	99.865	88.452

Building 182-N. The 182 Building model is shown in Figure 4.11 and consists of a two dimensional beam element finite element model. Separate models for the two horizontal directions and for the vertical direction were developed. Details of the structural model developed by Westinghouse are given in Reference 12.

Building 109-NT. The 109-NT building model is shown in Figure 4.12 and consists of a two dimensional beam element finite element model. Separate models for the two horizontal directions and for the vertical direction were developed. Details of the structural model developed by Westinghouse are given in Reference 12.

4.4.3.5 Soil-Structure Interaction Analysis

The probabilistic response analysis procedure performs a series of deterministic analyses, each simulating an earthquake occurrence, including variability in seismic input, soil-structure interaction, and in-structure representation. The seismic input variability is introduced by considering an ensemble of earthquake motions. For this study, the ten earthquake motions described earlier were used. A series of ten earthquake simulations, three components of motion, for each acceleration range were performed each using the identical free-field input motion as a starting point. Soil-structure interaction and structure response variability are introduced through a limited number of parameters, i.e., soil shear modulus, soil damping, structure frequency and structural modal damping. Variability in SSI was incorporated through modeling soil shear modulus and material damping as random variables with lognormal distributions, i.e., medians corresponding to the nominal values and coefficients of variation of 0.4 and 0.5, respectively. Variability in structure dynamic behavior was also modeled by treating structure frequency and modal damping as random variables. Parameter variations in each step of the response analysis were selected to represent random variability, not to include modeling uncertainty. The assumed parameter variability corresponds to that of the SSMRF (Ref. 22). The parameter values for each of the ten simulations were selected from the probability distributions by dividing the distributions into equally probable segments, sampling from each segment and combining the samples by a Latin hypercube experimental design. The responses calculated from the simulations are combined to estimate median responses conditional on the occurrence of an earthquake described by a particular hazard curve parameter, e.g., peak ground acceleration.

For all buildings, instructure spectra were calculated at 5 percent damping at the mass centroid of each floor elevation translational component for the ten input motions. The ten components then combined to form median spectra assuming a lognormal distribution.

PLOT21-INP.

ANSYS 4.3

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PLOT NO. 3

PREP7 ELEMENTS

MODE NUM

ZU-1

DIST-102 — ROOF

XF--133

VF-14.5

VRT0-2

— MEZZANINE

— GROUND FLOOR

— BASEMENT

Figure 4.12(a) Building 109-NT East-West Model

ANALYSIS 4.3
 JUL 7 1988
 17:15:47
 PLOT NO. 3
 PREP7 ELEMENTS
 NODE NUM
 ZU-1
 DIST-42.4
 XF-29.5
 VF-14.5

— ROOF

— MEZZANINE

— GROUND FLOOR

— BASEMENT

Figure 4.12(b) Building 109-NT North-South Model

Building 105-N/109-N/Process Area. Figures 4.13(a), through 4.13(c) are the median instructure response spectra for building 105-N at the 1 SSE level. Figures 4.14(a) and (b) are the median in-structure response spectra for Building 109-N. The free-field and foundation response spectra are overplotted with in-structure response.

Building 181-N Pumphouse. Figures 4.15(a) and (b) present median in-structure spectra plotted with the free-field and foundation reference point median spectra for the 1 SSE level. These spectra form the basis for determining the response of safety related components.

Building 182-N. Figure 4.16(a) through (c) present sample spectra computed for building 182-N at the 1 SSE level (for time history No. 1). Similar spectra for all ten time histories are presented in Reference 12. (Median spectra were not generated since it was the failure level of the building that was of primary interest).

Building 109-NT. Figures 4.17(a) through (c) present sample spectra computed for building 182-N at the 1 SSE level (for time history No. 1). Similar spectra for all ten time histories are presented in Reference 12. (Median spectra were not generated since it was the failure level of the building that was of primary interest).

4.4.4 Safety Related Component Responses

The in-structure spectra presented and discussed in the previous section are used to determine safety related component response. Assuming that the dynamic characteristics of a given component can be represented by a single dominant mode of vibration, the component response can be estimated by the spectral acceleration of the appropriate in-structure spectra at the frequency of the dominant mode. Thus, for each floor slab elevation in each building, one can identify the equipment of interest (at that location) and hence specify the frequencies (or frequency ranges) for which median spectral acceleration values are required in order to calculate the component failure probabilities. For example, Tables 4.10(a) through (c) give the spectral accelerations at various floor elevations and frequency ranges for Building 181-N, at the three ground motion levels. Similar tables for Buildings 105-N, 109-N and for the free field ground motion (for grade-mounted components) are given in Appendix D.

At each structural location, these spectral median responses are calculated over several frequency ranges using the spectra described in Section 4.3 above. These ranges span the estimated natural frequencies of the components housed at that location. The median response at

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 0'-0"
 el. 14'-6"
 el. 28'-3"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure 4.13(a) N-Reactor 105N Median In-Structure
 Response Acceleration Range 1

Figure 4.13(b) N-Reactor 105-N Median In-Structure
 Response Acceleration Range 1

Figure 4.13(c) N-Reactor 105-N Median In-Structure
Response Acceleration Range 1

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. -16'-0"
 el. 0'-0"
 el. 26'-0"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure 4.14(a) N-Reactor 109-N Median In-Structure
 Response Acceleration Range 1

Figure 4.14(b) N-Reactor Median 109-N In-Structure
Response Acceleration Range 1

Figure 4.15(a) N-Reactor 181-N Median In-Structure Response Acceleration Range 1

Figure 4.15(b) N-Reactor 181-N Median In-Structure Response Acceleration Range 1

Figure 4.16(a) N-Reactor Building 182-N In-Structure
 Response North-South (Acceleration
 Level 1, Time History No. 1)

BSHT NODE 45
 080 FLR NODE 42
 ROOF NODE 37

Figure 4.16(b) N-Reactor Building 182-N In-Structure
 Response East-West (Acceleration
 Level 1, Time History No. 1)

BSMT NODE 46
 FLR COL NODE 83
 MID ROOF NODE 85

Figure 4.16(c) N-Reactor Building 182-N In-Structure
 Response Vertical (Acceleration
 Level 1, Time History No. 1)

BSHT NODE 76
 GRD FLR NODE 70
 MEZZ NODE 84
 ROOF NODE 109

Figure 4.17(a) N-Reactor Building 109-NT In-Structure
 Response - North/South (Acceleration
 Level 1, Time History No. 1)

BSM1 NODE 350
 BRD FLR NODE 177
 MEZZ NODE 164
 ROOF NODE 152

Figure 4.17(b) N-Reactor Building 109-NT In-Structure
 Response - East/West (Acceleration
 Level 1, Time History No. 1)

BSMT NODE 76
 GRD FLR WALL NODE 70
 GRD FLR SPAN NODE 86
 MEZZ SPAN NODE 84

Figure 4.17(c) N-Reactor Building 109-NT In-Structure
 Response - Vertical (Acceleration
 Level 1, Time History No. 1)

Table 4.10
Building 181-N Median In-Structure Response (g)
(a) Acceleration Range 1

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
355'-0"	E-W	0.35	0.32	0.34	0.24	0.18
355'-0"	N-S	0.38	0.31	0.32	0.26	0.19
355'-0"	Ver	0.36	0.35	0.36	0.30	0.18
390'-0"	E-W	0.38	0.44	0.46	0.35	0.20
390'-0"	N-S	0.58	0.48	0.50	0.33	0.27
390'-0"	Ver	0.36	0.37	0.37	0.32	0.20
421'-6"	E-W	0.40	0.53	0.54	0.44	0.23
421'-6"	N-S	0.67	0.69	0.73	0.48	0.32
421'-6"	Ver	0.37	0.37	0.38	0.33	0.21
439'-0"	E-W	0.46	1.00	1.01	0.98	0.37
439'-0"	N-S	0.98	1.56	1.69	0.96	0.59
439'-0"	Ver	0.37	0.39	0.39	0.36	0.25
Crane	E-W	0.92	0.59	0.60	0.45	0.38
Crane	N-S	1.02	0.56	0.56	0.49	0.45

Table 4.10
Building 181-N Median In-Structure Response (g)
(b) Acceleration Range 2

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
355'-0"	E-W	0.72	0.64	0.67	0.49	0.36
355'-0"	N-S	0.76	0.61	0.63	0.51	0.36
355'-0"	Ver	0.73	0.69	0.70	0.60	0.36
390'-0"	E-W	0.80	0.88	0.91	0.66	0.40
390'-0"	N-S	1.13	0.88	0.91	0.64	0.50
390'-0"	Ver	0.74	0.72	0.73	0.63	0.37
421'-6"	E-W	0.85	1.09	1.11	0.86	0.45
421'-6"	N-S	1.31	1.25	1.31	0.89	0.60
421'-6"	Ver	0.74	0.73	0.74	0.65	0.39
439'-0"	E-W	0.99	1.99	2.01	1.85	0.73
439'-0"	N-S	1.87	2.66	2.84	1.65	1.06
439'-0"	Ver	0.75	0.77	0.77	0.71	0.43
Crane	E-W	1.88	1.20	1.22	0.93	0.78
Crane	N-S	1.91	1.03	1.04	0.90	0.82

Table 4.10
 Building 181-N Median In-Structure Response (g)
 (c) Acceleration Range 3

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
355'-0"	E-W	1.17	0.96	0.98	0.74	0.53
355'-0"	N-S	1.16	0.94	0.95	0.80	0.54
355'-0"	Ver	1.10	1.01	1.02	0.86	0.52
390'-0"	E-W	1.33	1.28	1.33	0.91	0.61
390'-0"	N-S	1.79	1.30	1.34	0.95	0.73
390'-0"	Ver	1.11	1.04	1.05	0.90	0.54
421'-6"	E-W	1.45	1.63	1.69	1.20	0.68
421'-6"	N-S	2.12	1.85	1.90	1.37	0.91
421'-6"	Ver	1.12	1.06	1.07	0.93	0.56
439'-0"	E-W	1.70	2.99	3.08	2.44	1.07
439'-0"	N-S	3.09	3.83	4.03	2.52	1.64
439'-0"	Ver	1.14	1.13	1.13	1.02	0.60
Crane	E-W	3.13	1.94	1.98	1.48	1.23
Crane	N-S	2.78	1.47	1.47	1.26	1.16

specific frequencies is calculated from the ten values given by the probabilistic response analysis assuming a lognormal distribution. The median response over a frequency range is based on an average over the frequency interval.

Correction for Imbedment

Specification of the control point at the elevation of the base of the foundation for 105-N/109-N/process area instead of at the grade introduced some conservatism in structural response for these structures.

A previous study (Ref. 23) was conducted to evaluate the effects of control point location on the response of buildings 105-N and 109-N. Results from the study in the form of the ratio between horizontal in-structure response spectra (control point at 454 feet, control point at 432 feet) are given in Figures 4.18 through 4.21 for buildings 105-N and 109-N, for an input motion of 0.25g. Therefore, the component corrected response is the product of the in-structure response and soil replacement factor at the appropriate elevation and frequency range. Tables 4.11 and 4.12 show the median ratio of response over the same frequency ranges as the systems component responses were derived.

The soil replacement factors are applicable for acceleration range 1. Degradation of soil properties and the rigid body response of the buildings 105-N and 109-N causes a shift in soil-structure system frequency and less amplification of in-structure response with the higher acceleration ranges, and, hence, were not used for these analyses.

4.4.4.1 Responses in Terms of Peak Ground Acceleration

The response spectra given in Figures 4.13 through 4.17 and Appendix D constitute the complete set of responses calculated for the N-Reactor structures. Only a subset of these were required for the component failure calculations. These spectral acceleration responses are given on Table 4.10 and in Appendix D at three peak ground acceleration values (0.25 g, 0.50 g, and 0.75 g) for all structures and floor levels of interest. One can directly interpolate between these three values to obtain any specified response at any arbitrary value of peak ground acceleration.

However, a more direct approach which greatly simplifies computation of the component failure probabilities is to compute the average ratio between the median PGA and the median response spectral acceleration at each specified component location. Figures 4.22 through 4.27 are plots of the response location accelerations in each building (at various building elevations) versus PGA. It can be seen that a linear relation exists up to free field accelerations of 0.75 g or greater. Furthermore, for those curves which show significant non-linearity at

Figure 4.18 Building 105-N Effect of Embedment on In-Structure Response East/West

Legend:

el. -22'-0"
 el. 0'-0"
 el. 44'-0"

Notes:

Spectra for ratio calculated at 5% damping
 Structure response calculated for .25 g input motion

Figure 4.19 Building 105-N Effect of Embedment
 on In-Structure Response North/South

Legend:

el. -22'-0"
 el. 0'-0"
 el. 44'-0"

Notes:

Spectra for ratio calculated at 5% damping
 Structure response calculated for .25 g input motion

Figure 4.20 Building 109-N Effect of Embedment
 on In-Structure Response East/West

Legend:

el. -22'-0"
 el. 0'-0"
 el. 44'-0"

Notes:

Spectra for ratio calculated at 5% damping
 Structure response calculated for .25 g input motion

Figure 4.21 Building 109-N Effect of Embedment on
 In-Structure Response North/South

Table 4.11

Building 105-N Soil Replacement Factors
Acceleration Range 1

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	0.68	0.78	0.80	0.78	0.78
-22'-0"	N-S	0.78	0.77	0.77	0.74	0.80
0'-6"	E-W	0.67	0.70	0.70	0.73	0.75
0'-6"	N-S	0.70	0.77	0.76	0.77	0.80
44'-0	E-W	0.66	0.65	0.65	0.68	0.70
44'-0"	N-S	0.66	0.68	0.68	0.70	0.69

Table 4.12

Building 109-N Soil Replacement Factors
Acceleration Range 1

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	0.72	0.92	0.97	0.84	0.75
-22'-0"	N-S	0.75	0.82	0.87	0.73	0.76
0'-0"	E-W	0.73	0.91	0.94	0.84	0.75
0'-0"	N-S	0.71	0.80	0.84	0.76	0.73
44'-0	E-W	0.73	0.93	0.96	0.89	0.75
44'-0"	N-S	0.71	0.86	0.88	0.83	0.73

Figure 4.22 Spectral Responses vs. PGA for the Free Field

Figure 4.23 Spectral Responses vs PGA for Building 105-N

Figure 4.24 Spectral Responses vs PGA for Building 109-N

Figure 4.25 Spectral Responses vs PGA for Building 109-NT

Figure 4.26 Spectral Responses vs PGA for Building 181-N

Figure 4.27 Spectral Responses vs PGA for Building 182-N

higher acceleration levels, the linear relation provides a conservative estimate of the local response. Inasmuch as the bulk of the risk at N Reactor is found to be due to earthquake levels in the 1 to 3 SSE level (0.25g to 0.75g), it can be seen that relating local response to peak ground acceleration in a linear fashion is appropriate for the N Reactor seismic analysis.

From these tables, ratios between the various responses and PGA were determined, as listed in Table 4.13. For each frequency (or frequency range) the response multiple is defined as the ratio of the maximum horizontal response at 0.25g to the free field response at 0.25g. The cross-reference file (XREF) given in Appendix E lists all the basic component failure events, their locations, response numbers and fragility categories. These two tables can be used in conjunction with tables of the site-specific and generic component fragility failure values to obtain any desired component failure probability directly (as described in Section 4.6.).

4.4.4.2 Variability in Response

Variability in responses (floor and spectral accelerations) was assigned based on SSMRP results (Ref. 22). Confidence bounds were computed for the final core damage frequencies using both random (irreducible) and systematic (modeling) uncertainties. The uncertainties (expressed as standard deviations of the logarithms of the responses) are shown below:

<u>Quantity</u>	<u>Random</u>	<u>Modeling</u>
Peak Ground Acceleration	0.25	See response
Floor Zero Period Acceleration	0.35	table in
Floor Spectral Acceleration	0.45	Appendix E

4.4.4.3 Correlation

In computing the correlation between component failures (in order to quantify the cutsets) it is necessary to consider correlations both in the responses and in the fragilities of each component. Inasmuch as there are no data as yet which prove or disprove correlation between fragilities, the fragility correlations between like components were taken as zero. The correlation between responses is assigned according to the rules on Table 4.14.

Finally, the correlation coefficient between any two component failures is computed from the expression

$$\rho = \frac{\beta_{R1}\beta_{R2}}{\sqrt{\beta_{R1}^2 + \beta_{F1}^2} \sqrt{\beta_{R2}^2 + \beta_{F2}^2}} \rho_{R1R2} + \frac{\beta_{F1}\beta_{F2}}{\sqrt{\beta_{R1}^2 + \beta_{F1}^2} \sqrt{\beta_{R2}^2 + \beta_{F2}^2}} \rho_{F1F2}$$

Table 4.13

N Reactor Seismic Response Locations

<u>No.</u>	<u>Factor Times PGA</u>	<u>Response beta-r</u>	<u>Response beta-u</u>	<u>Building, Elevation and Frequency</u>
1	1.00	0.25	.25	FREE-FIELD ZPA
2	2.32	0.45	.25	2-5 HZ
3	1.80	0.35	.25	5-10
4	2.60	0.35	.25	3
5	1.12	0.35	.25	105-N (-) 22'-0" ZPA
6	1.72	0.45	.25	5-10 HZ
7	1.16	0.35	.25	0'-0" ZPA
8	1.72	0.45	.25	5-10 HZ
9	1.20	0.35	.25	14'-6" ZPA
10	1.24	0.35	.25	28'-3" ZPA
11	1.24	0.35	.25	40'-0" ZPA
12	1.28	0.35	.25	51'-0" ZPA
13	1.32	0.35	.25	60'-6" ZPA
14	1.08	0.35	.25	109-N (-) 16'-0" ZPA
15	3.16	0.45	.25	2-5 HZ
16	3.60	0.45	.25	3 Hz
17	1.72	0.45	.25	5-10 HZ
18	1.12	0.35	.25	0'-0" ZPA
19	2.92	0.45	.25	2-5 Hz
20	3.66	0.45	.25	3 Hz
21	1.76	0.45	.25	5-10 HZ
22	1.20	0.35	.25	26'-0" ZPA
23	0.88	0.35	.25	109-NT (-) 16'-0" ZPA
24	2.12	0.45	.25	5-10 HZ
25	1.24	0.35	.25	0'-0" ZPA
26	2.12	0.45	.25	5-10 HZ
27	2.15	0.35	.25	15'-3" ZPA
28	1.88	0.35	.25	36'-8" ZPA
29	0.80	0.35	.25	151-N (-) 16'-0", (-) 24'-0" ZPA

Table 4.13 (Cont'd)

N Reactor Seismic Response Locations

<u>No.</u>	<u>Factor</u> <u>Times PGA</u>	<u>Response</u> <u>beta-r</u>	<u>Response</u> <u>beta-u</u>	<u>Building, Elevation</u> <u>and Frequency</u>	
30	1.64	0.45	.25	153-N	5-10 HZ
31	0.84	0.35	.25	182-N	0'-0" ZPA
32	2.08	0.45	.25		5-10HZ
33	1.96	0.35	.25		16'-0" ZPA
34	0.72	0.35	.25	181-N	360'-0" ZPA
35	1.50	0.45	.25		3-7 HZ
36	1.28	0.45	.25		5-10 HZ
37	1.08	0.35	.25		390'-0" ZPA
38	2.48	0.45	.25		3-7 HZ
39	2.02	0.45	.25		5-10 HZ
40	1.28	0.35	.25		421'-0" ZPA
41	3.04	0.45	.25		3-7 HZ
42	2.88	0.45	.25		5-10 HZ
43	2.40	0.35	.25		440'-0" ZPA
44	6.68	0.45	.25		5-10 HZ
45	8.16	0.45	.25		6 HZ N-S COMPONENT
46	1.24	0.35	.25	184-N	0'-0" ZPA (SAME AS 109-NT)
47	2.12	0.45	.25		5-10 HZ
48	1.24	0.35	.25	109-N	39'-6" ZPA
49	1.28	0.35	.25		60'-6" ZPA
50	1.36	0.35	.25	184-N	15'-3" ZPA
51	1.88	0.35	.25		36'-8" ZPA
52	2.80	0.45	.25		15'-3" 5-10 HZ
53	1.56	0.45	.25	105-N	(-)16'-0" 5-10 HZ (ROOM 6)

Table 4.14

Rules for Assigning Response Correlation

ρ_{R1R2}

-
1. Components on the same floor slab, and sensitive to the same spectral frequency range (i.e., ZPA, 5-10 Hz, or 10-15 Hz) will be assigned response correlation = 1.0.
 2. Components on the same floor slab, sensitive to different ranges of spectral acceleration will be assigned response correlation = 0.5.
 3. Components on different floor slabs (but in the same building) and sensitive to the same spectral frequency range (ZPA, 5-10 Hz or 10-15 Hz) will be assigned response correlation = 0.75.
 4. Components on the ground surface (outside tanks, etc.) shall be treated as if they were on the grade floor of an adjacent building, and then use rules 1 to 3 as appropriate.
 5. "Ganged" valve configurations (either parallel or series) will have response correlation = 1.0.
 6. All other configurations will have response correlation = 0.
-

in which

ρ - correlation coefficient between the failure of components 1 and 2

β_{R1}, β_{R2} - standard deviation of the logarithms of the responses of components 1 and 2

β_{F1}, β_{F2} - standard deviations of the logarithms of the fragilities of components 1 and 2

ρ_{R1R2} - correlation coefficient between responses of components 1 and 2

ρ_{F1F2} - correlation coefficient between the fragilities of components 1 and 2

This relation shows that the correlation between the failure of components 1 and 2 depends not only on the correlations between the respective responses and the respective fragilities, but also on the variances in these responses and fragilities.

A review of the response correlation table in conjunction with the fact that fragility correlations are taken as zero allowed screening of the components for those differing components which might be assigned correlation. For these identical components in a common location, the correlation is specified in the cross reference file (XREF in Appendix E) by the number in the column NCORR. For these components, a failure correlation of 0.5 (based on a response correlation of 1.0 and a fragility correlation of zero) was assigned for each correlated group in the cutsets.

4.4.5 Reactor Core Seismic Response Evaluation

4.4.5.1 Introduction

As part of the evaluation of the structural seismic response and fragility of the N-Reactor buildings, a detailed evaluation of the reactor core and enclosing primary shield was made. This was done to develop failure criteria for the process tubes taking into account the following unique factors:

- a. Horizontal Pressures,
- b. Graphite core construction with potential for lateral sliding motion,
- c. Aging effects for both zircalloy tubes and graphite stack,
- d. Current configuration involving some failed graphite "keys", and
- e. Potential for impact between graphite stock and primary shield wall.

Many of these aspects had been raised in the 1987 National Academy of Sciences review of DOE Productions Reactors as contained in Reference 23.

The N-Reactor core is a structure 33 feet wide by 33 feet 4 inches high by 39 feet 5 inches long. It consists of 89 layers of interconnected keyed graphite blocks, with roughly 2000 keys per block (see Figure 4.28). The reactor core is free standing, uncoupled from building 105-N. It is surrounded by the primary shield, which is a high density concrete box structure 42 feet 6 inches high by 41 feet 1 inches long by 34 feet 2 inches long; its walls are enclosed in 1.25-in. steel plates, used as the original form. The front and back faces of this box structure, called inlet and outlet faces, have a large wall supported on rollers at the base and unsupported in their own plane on the remaining three sides except for a small lug at the top; out of plane displacements are limited by the primary shield. Process tubes and cooling tubes extend from the reactor through the primary shield and the inlet and outlet shields. There are 1003 process tubes running in the N-S direction, 640 graphite cooling tubes and 84 horizontal control rods running in the E-W direction. A 1.5-in. gap separates the core and the primary shield on all four sides. Both the inlet and the outlet shields are separated from the primary shield by a 0.75-in. gap on each vertical face of the wall. Finally, the primary shield is separated from the 105-N structure by a 5-in. gap in the N-S direction at the base, and by a 2-in. gap in the E-W direction throughout its height.

A nonlinear seismic analysis of the reactor core was performed to assess the structural fragility of the major substructures forming the core, and their relative motion. Responses were calculated for all three acceleration ranges, i.e., 1 SSE, 2 SSE and 3 SSE. The input time histories were the calculated in-structure response of the 105-N/109-N/process area at the base of the process area (Section 4.3.5). Each horizontal direction is analyzed separately.

4.4.5.2 Model Development

4.4.5.2.1 Substructures

The core analysis considers four substructures for the process area:

- a. The graphite core
- b. The primary shield
- c. The inlet and outlet shields
- d. The 105-N structure

The following models were developed for these substructures:

- a. The core model is represented by one (N-S) or two (E-W) degrees of freedom. The mass and lateral stiffness of the core includes both the fuel zone and reflector zone. The fuel and reflector

Figure 4.28 N-Reactor Process Area Configuration

zones are connected at the base and top of the core. Therefore, a single model is considered adequate to represent the behavior of the core under seismic loading,

- b. The primary shield model is represented by one (N-S) or two (E-W) degrees of freedom. The models are reduced versions of more detailed representations of the primary shield's dynamic behavior used in its fragility assessment, but still capture the major behavioral aspects of the structure,
- c. The inlet/outlet Shield model is represented by two degrees of freedom in the E-W direction. In the north-south direction, the inlet and outlet shields are incorporated into the primary shield model, and
- d. The 105-N structure was modeled as a fixed rigid body, i.e., as a boundary for impact.

4.4.5.2.2 N-S Model

The analysis in the N-S direction included models of the core, the primary shield and the 105-N structure. A series of preliminary studies was performed to derive simplified models of the core and the primary shield, yet retaining the important aspects of their response. In the preliminary phase, the core model and the primary shield model were analyzed separately and uncoupled. The core model included the sliding phenomenon between each layer and between the bottom layer and the base. Energy dissipation was treated through hysteresis loops of the friction elements. The core properties were derived from References 10 and 20.

The primary shield model included undamped elastic deformation of the structure and sliding at the base. This representation accounted for the fact that the primary shield base plates are grouted into a pocket in the foundation. These embedded steel shapes will function as lugs against E-W seismic forces; however, resistance to N-S forces will be low as the seismic loads are unlikely to be carried around the corners to the base plates for the inlet and outlet shields, hence permitting sliding to occur. An average effective coefficient of friction for steel on grout of 0.60 is estimated from Reference 25.

Vertical acceleration was conservatively treated as a statically applied acceleration acting in the worst case direction, i.e., in the upward vertical direction to reduce the friction force. The peak vertical acceleration was used for this purpose.

These preliminary studies showed that both structures behave as rigid bodies sliding at their base; model discretization was not important as the response was dominated by friction at the base. The studies also showed that the two structures impacted between the 1 SSE and 2 SSE levels. Sensitivity studies were performed to evaluate the importance

of base rocking, variation in model properties and variation in the input time history. Only the input time history was found to be of noticeable importance on the overall response.

A best estimate model was then assembled. The N-S model is shown in Figure 4.29. The core and the primary shield were each represented as a sliding rigid body by a one lumped mass model, connected to the base by a sliding element; a linear elastic-perfectly plastic force-displacement relationship was used. The elastic stiffness of the sliding element was calculated to match the elastic natural frequency of the structure. From Reference 10, the fundamental frequency of the core was calculated as 16.6 Hz and per Reference 20, the elastic frequency of the primary shield was taken as 19.0 Hz.

The friction force of the core sliding element corresponded to the friction between the bottom graphite layer and the steel base plate (coefficient of friction is taken as 0.21). The friction force of the primary shield sliding element corresponds to a coefficient of friction of steel on grout, taken as 0.6 per Reference 25 at the base of the shield.

No connectivity through the cooling tubes was included, as the relative stiffness of these tubes is small compared to the elastic stiffness of both the core and the primary shield. Viscous material damping is applied when the structures are not sliding (elastic deformation); a damping ratio of 5 percent of critical was assumed for both models. Note, such elastic deformation induces relatively low stresses in the core and the primary shield structures. The two structures are connected by a gap element which is activated only when one structure impacts the other. The 105-N structure is modeled as a fixed rigid body relative to the core and primary shield due to its large relative mass and stiffness. The primary shield model and the 105-N structure are separated by a gap element.

4.4.5.2.3 E-W Model

All four substructures are represented in this model. Preliminary studies were performed to derive simplified best estimate models. In these preliminary studies, the primary shield was modeled as two non-linear frames slaved at the roof; one frame represented the inlet/outlet frames (north and south faces) and the second represented the left and right faces called left/right frame (see Figure 4.30). Stiffness degradation of the concrete due to cracking and additional seismic capacity from the exterior steel plates, were modeled. The frames were pin-supported at the base because the reinforcement is not doweled to the foundation. The core model for these preliminary studies was similar to the model for the N-S studies, even though the core is keyed at the base in the E-W direction. The inlet/outlet shields were represented by their elastic shear stiffness. They are keyed to the primary shield by a single lug at the top. This connection detail was

N-S Model

LEGEND

- Lumped Mass
- Non linear horizontal sliding spring element
- Gap element

Figure 4.29 North-South Model

Table 4.15

N-S Response Peak Displacements Relative to
Base of Structure

	Max. Core Displacement (ft)	Max. Shield Displacement (ft)	Max. Differential Displacement (ft)
1 SSE			
t1	.0264	.0006	.0268
t2	.0243	.0006	.0245
t3	.0594	.0007	.0595
t4	.0599	.0008	.0603
t5	.1410	.0159	.126
t6	.0764	.0006	.0768
t7	.0779	.0005	.0779
t8	.1290	.0081	.126
t9	.0264	.0006	.0265
t10	.0625	.0006	.0627
Average	.0683	.0029	.0667
Stand Dev.	.0384	.0049	.0352
2 SSE			
t1	.2030	.0874	.126
t2	.1590	.0409	.126
t3	.1890	.0940	.127
t4	.2560	.1310	.127
t5	.5380	.4230	.130
t6	.1280	.0682	.125
t7	.4890	.3640	.128
t8	.4520	.3470	.127
t9	.1630	.0474	.126
t10	.3050	.2270	.127
Average	.2882	.1830	.1269
Stand Dev.	.1435	.1381	.0013

Table 4.15 (cont'd)

N-S Response Peak Displacements Relative to
Base of Structure (Concluded)

	Max. Core Displacement (ft)	Max. Shield Displacement (ft)	Max. Differential Displacement (ft)
3 SSE			
t1	.5430	.4180	.127
t2	.2280	.1210	.127
t3	.5360	.4190	.127
t4	.4610	.3710	.127
t5	.5610	.4320	.138
t6	.2740	.1620	.127
t7	.5450	.4270	.134
t8	.5440	.4230	.130
t9	.3030	.1780	.127
t10	.5420	.4220	.131
Average	.4537	.3373	.1295
Stand Dev.	.1251	.1220	.0036

SHIELD-1 50007.01

Figure 4.30 Primary Shield E-W Model for Preliminary Analysis

estimated sufficient to restrain any thermal movements, but was estimated to fail under relatively low seismic loading. Therefore the connection between the inlet/outlet shields and the primary shield models is represented only by a friction element at the base of the inlet/outlet shields, with a small coefficient of friction (0.08).

These studies showed that the primary shield response was highly non-linear even at 1 SSE, as it behaved as a flexible moment frame. The studies also showed that the connectivity of the inlet/outlet shields and the core models through the process tubes was an important factor in the system response. Furthermore, the primary shield impacted the inlet/outlet shields at excitations smaller than the 1 SSE level.

A best estimate model was then assembled. The E-W model is shown in Figure 4.31. Each substructure was represented by a two lumped mass model.

The core model included the sliding phenomenon between each layer and between the bottom layer and the base of the structure, as well as the resistance of relative motion between layers due to the keys. This is of particular importance since the core is keyed at its base in this direction. In the N-S direction, core response is basically rigid body motion with sliding at its base. In the E-W direction, relative motion throughout the core is expected due to keying at its base, and the effect of the process tubes on its response. The core properties were derived from References 10 and 20. From Reference 10, the core elastic frequency was calculated as about 17 Hz as before. For each stiffness model, the properties of the sliding element and of the key spring element were such that the equivalent element had a tri-linear force-displacement relationship, shown in Figure 4.32. The stiffness up to the friction force corresponds to the linear elastic stiffness; the strain hardening stiffness value corresponds to the tangent stiffness approximation of the key action as evaluated in Reference 20.

The primary shield model was reduced to a non-linear shear beam representation, including a linear elastic-perfectly plastic force-displacement relationship. The primary shield model properties were derived from the preliminary analyses of the more refined model and capture the important behavioral aspects of the structure. The elastic frequency of the primary shield was calculated as 2 Hz.

The inlet/outlet shields model included linear shear deformation and sliding at the base, with a coefficient of friction of 0.08. The properties were derived from Reference 20.

Vertical acceleration was treated as a statically applied load, by applying the peak vertical acceleration in the upward vertical direction.

The core and the inlet/outlet models are connected through the process tubes. Gap elements connect on one hand the core and the primary

E-W Model

LEGEND

- Lumped Mass
- Non linear horizontal sliding spring element
- Non linear shear beam element
- Linear elastic shear beam element
- Non linear key spring element
- Linear elastic process tubes spring element
- Gap element

Figure 4.31 East-West Model

Figure 4.32 Force-Displacement Relationship for Core E-W Model

shield, and on the other the primary shield and the inlet/outlet shields. The 105-N structure is modeled as a fixed rigid body relative to the core and primary shield due to its large relative mass and stiffness. It is connected to the primary shield by a third pair of gap elements.

Viscous material damping is included for all three models and is taken as 5 percent of critical at all three acceleration ranges. Material damping is also included for the process tubes and is taken (per Reference 26) as 2 percent of critical.

4.4.5.2.4 Gap element

A gap element was used to model the dynamic behavior of two adjacent structures separated by a gap of finite dimension. The force-deformation relationship of the gap element is shown in Figure 4.33.

The stiffness of the gap element is calculated using Boussineq's solution of a concentrated load acting on an elastic half space (Ref. 27).

Impact damping is also introduced in the form of a viscous damping for the gap element. It represents the amount of energy lost during impact. When a simple model such as in Figure 4.34 is used, the damping ratio and the coefficient of restitution can be correlated (Ref. 28). An impact damping ratio of 10 percent of critical is included in the analysis for each gap element, corresponding to a coefficient of restitution of roughly 73 percent.

4.4.5.3 Seismic Analysis of the Core

Ten non linear time history analyses were performed for each acceleration range in each horizontal direction (see Section 4.3).

4.4.5.3.1 N-S Nonlinear Seismic Response

The N-S core and primary shield responses are dominated by sliding at the base, and elastic deformation is very small. Peak displacements of both structures relative to their base and relative displacement between the two structures are of interest. A summary of the results are presented in Table 4.15.

At the 1 SSE level, the core is sliding and displacements are sensitive to the input time history, as the peak values vary between 0.3 inches and 1.7 inches. The primary shield slides for two simulations out of ten. Its displacement is relatively small when the maximum shear force is smaller than the friction at the base (peak of 0.01 inches). For time histories 5 and 8, for which the primary shield slides, its displacement is respectively 0.2 and 0.1 inches. For these simulations, the gap between the core and the shield is predicted to close as the structures impact each other, whereas for the remaining eight others, the gap is not predicted to close.

Figure 4.33 Force-Displacement Relationship for Gap Elements

Figure 4.34 Impact Damping Modelization

Before impact between the two structures occurs, both responses are independent and uncoupled. When impact occurs, the response of the two structures are coupled. The core traveling faster, impacts the primary shield, transferring energy by trying to push it further. System energy is dissipated by friction at the base of the core and the primary shield, by elastic viscous damping, and by impact damping between the two structures. At the 2 SSE level, the core and the primary shield impact each other for all ten simulations. For simulation 5, the gap between the primary shield and the 105-N structure is also predicted to close and these two structures impact. At the 3 SSE level, the gap between the core and the shield closes for all ten simulations and the gap between the shield and the 105-N structure is predicted to close for six out of the ten simulations.

When the core impacts the primary shield, the core displacement is limited by that of the shield; therefore the peak differential displacement between the two structures is about the gap, i.e., 1.5 inches. Similarly, when the primary shield impacts the 105-N structure, the 105-N structure acts as a stop for the primary shield limiting the peak shield displacement to approximately the gap, i.e., 5 inches.

The results show, as expected, that the response of both structures is sensitive to the input motion; for the same excitation level, the PGA is roughly the same for all ten excitations. This is characteristic of non-linear analysis where the structural frequency varies with time. This sensitivity is emphasized due to impact between structures.

In-structural response spectra were calculated for damping ratios of 2 percent and 5 percent of critical. Figures 1 through 12 in Appendix F are examples of core and shield response for two input motions at all three acceleration ranges. When no impact occurs, the primary shield response spectra have large peaks at roughly 2.5 Hz and 3.5 Hz; the core response spectra have large peaks at frequencies ranging from 2 to 5 Hz. The response spectra for both structures are somewhat similar in shape and are also similar to the input response spectra. This is expected because when the structures slide, little or no dynamic amplification occurs.

When the gap between the two structures closes, the sudden increase in stiffness due to the impact idealization, creates artificially high spectral accelerations at high frequencies. (See the figures in Appendix D for the frequencies above 10 Hz.). These unrealistic spectral accelerations are simply an artifact of the representation used in the analysis and should be used with caution.

4.4.5.3.2 E-W Nonlinear Seismic Response

The E-W primary shield response is dominated by the non-linear shear beam behavior, with relatively large differential displacement between

the top and bottom nodes. This is due to the representation of the primary shield; a more refined model would show that the shield actually perform as moment frames.

The inlet/outlet shields behave as rigid bodies sliding. The gap with the primary shield closes at all three levels of excitation, for all ten simulations. Since this gap is small, and the friction force at the base of the inlet/outlet shields is relatively small, and also the primary shield is very flexible, the inlet/outlet shields are bouncing back and forth against the primary shield, creating a pounding effect. When the gap between the inlet/outlet shield and the primary shield closes, one source of energy dissipation is through non-linear behavior of the primary shield. Thus, the inlet/outlet shields have a tendency to push the already flexible primary shield when they impact it. Once the primary shield moves, the inlet/outlet shields can also displace further. The core response is characterized by an initial sliding, followed by a non-linear shear beam behavior, with sizeable differential displacement between the top and the bottom. The core and the inlet/outlet shields' responses are coupled through the process tubes.

Peak displacements of all three structures relative to their base and to each other are of interest. A summary of the results is presented in Tables 4.16 and 4.17.

At 1 SSE, the gap between the inlet/outlet shields and the primary shield is predicted to close at the top for all simulations and is predicted to close five times out of ten at the bottom. The gap elements between the inlet/outlet and primary shields, represent the gaps on each side of the inlet/outlet shields. Gap closure at the top and bottom nodes probably occur at opposite sides of the inlet/outlet shield as shown in Figure 4.35. The gap between the core and the primary shield as well as the one between the primary shield and the 105-N structure are not predicted to close. The core displacement relative to its base is about 0.5 inches at 5.4 feet above its base and is about 0.7 inches at 35.3 ft. The primary shield displacement varies from .008 inches to .04 inches at 5.4 feet and from 0.8 inches to 1.5 inches at 35.3 feet above its base. The displacement of the inlet/outlet shield is restrained by the primary shield, as the differential displacement between the two structures is roughly 0.75 inches, the size of the gap separating them. At 2 SSE, the gap between the inlet/outlet shield and the primary shield closes both at the top and bottom for all ten simulations. The core and the primary shield never impact but the primary shield impacts the 105-N structure at 35.3 feet above its base for four simulations out of ten. The core displaces on average 1 inches and on average of 1.6 inches at 5.4 feet and 35.3 feet above its base respectively. The primary shield displaces about 0.5 inches and about 1.9 inches at 5.4 feet and 35.3 feet above its base respectively. At 3 SSE, the inlet/outlet shield always impacts the primary shield both at the top and at the bottom. The primary shield impacts Building 105-N at 35.3 feet above the base of the primary shield

Table 4.16

E-W Response Peak Displacements Relative to
Base of Structure

Elevation	Max Core Displacement (ft)		Max Primary Shield Displacement (ft)		Max I/O Shield Displacement (ft)	
	Bottom (5.43')	Top (35.3')	Bottom (5.43')	Top (35.3')	Bottom (5.43')	Top (35.3')
1 SSE						
t1	.0460	.0778	.0033	.117	.0653	.0659
t2	.0184	.0358	.0010	.0950	.0431	.0429
t3	.0662	.0799	.0034	.104	.0659	.0662
t4	.0557	.0717	.0015	.0925	.0640	.0642
t5	.0493	.0677	.0008	.0656	.0600	.0601
t6	.0508	.0747	.0011	.115	.0591	.0591
t7	.0535	.0725	.0011	.107	.0632	.0633
t8	.0281	.0386	.0009	.0986	.0508	.0511
t9	.0162	.0212	.0007	.0679	.0266	.0266
t10	.0667	.0732	.0049	.124	.0675	.0676
Average	.0451	.0613	.0019	.0987	.0566	.0567
Stand Dev.	.0173	.0200	.0014	.0185	.0123	.0124
2 SSE						
t1	.126	.175	.0668	.169	.129	.129
t2	.0527	.109	.0176	.141	.0802	.0805
t3	.117	.146	.0633	.167	.126	.126
t4	.0811	.122	.0290	.146	.0915	.0921
t5	.0851	.140	.0346	.156	.0971	.0976
t6	.107	.167	.0519	.167	.114	.115
t7	.109	.153	.0403	.166	.105	.105
t8	.0580	.113	.0144	.137	.0769	.0775
t9	.0575	.0846	.0032	.127	.0657	.0659
t10	.141	.163	.0677	.169	.130	.131
Average	.0934	.1373	.0391	.1545	.1015	.1020
Stand Dev.	.0296	.0277	.0220	.0148	.0219	.0219

Table 4.16 (cont'd)

E-W Response Peak Displacements Relative to
Base of Structure (Concluded)

Elevation	Max Core Displacement (ft)		Max Primary Shield Displacement (ft)		Max I/O Shield Displacement (ft)	
	Bottom	Top	Bottom	Top	Bottom	Top
	(5.43')	(35.3')	(5.43')	(35.3')	(5.43')	(35.3')
3 SSE						
t1	.160	.225	.1068	.170	.168	.169
t2	.0543	.112	.0193	.143	.0818	.0822
t3	.173	.222	.0961	.171	.159	.159
t4	.0815	.127	.0248	.147	.0873	.0878
t5	.145	.217	.0902	.168	.1531	.153
t6	.113	.172	.0493	.168	.112	.113
t7	.190	.221	.1157	.169	.177	.178
t8	.0587	.0976	.0045	.130	.0670	.0678
t9	.0809	.120	.0233	.147	.0858	.0863
t10	.173	.216	.0937	.170	.156	.157
Average	.1229	.1760	.0622	.1583	.1247	.1253
Stand Dev.	.0488	.0543	.0398	.0142	.0390	.0398

Table 4.17

E-W Response Peak Differential Displacement Between Structures

Elevation	Max Diff Displacement Between C & P.S (ft)		Max Diff Displacement Between C & I/O (ft)		Max Diff Displacement Between P.S. & I/O (ft)	
	Bottom	Top	Bottom	Top	Bottom	Top
	(5.43')	(35.3')	(5.43')	(35.3')	(5.43')	(35.3')
1 SSE						
t1	.0442	.0885	.0430	.0339	.0629	.0632
t2	.0183	.0795	.0381	.0311	.0423	.0633
t3	.0647	.0884	.0356	.0340	.0629	.0634
t4	.0552	.0826	.0433	.0313	.0627	.0633
t5	.0494	.0785	.0267	.0216	.0597	.0629
t6	.0507	.0854	.0449	.0307	.0586	.0633
t7	.0536	.0842	.0281	.0309	.0627	.0632
t8	.0282	.0717	.0301	.0234	.0501	.0633
t9	.0165	.0611	.0174	.0151	.0269	.0630
t10	.0636	.0985	.0376	.0383	.0630	.0632
Average	.0444	.0818	.0345	.0290	.0552	.0632
Stand Dev.	.0166	.0097	.0083	.0066	.0115	.0001
2 SSE						
t1	.0665	.1048	.0590	.0628	.0638	.0636
t2	.0438	.0841	.0555	.0359	.0634	.0634
t3	.0726	.0925	.0518	.0484	.0635	.0638
t4	.0671	.0924	.0573	.0447	.0630	.0635
t5	.0647	.0883	.0443	.0447	.0631	.0632
t6	.0663	.0966	.0614	.0527	.0635	.0634
t7	.0720	.1037	.0527	.0515	.0633	.0633
t8	.0449	.0780	.0620	.0420	.0630	.0633
t9	.0562	.0842	.0397	.0312	.0629	.0632
t10	.0743	.0997	.0556	.0494	.0633	.0638
Average	.0628	.0924	.0539	.0463	.0633	.0635
Stand Dev.	.0104	.0084	.0068	.0085	.0003	.0002

Table 4.17 (cont'd)

E-W Response Peak Differential Displacement Between
Structures (Concluded)

Elevation	Max Diff Displacement Between C & P.S. (ft)		Max Diff Displacement Between C & I/O (ft)		Max Diff Displacement Between P.S. & I/O (ft)	
	Bottom	Top	Bottom	Top	Bottom	Top
	(5.43')	(35.3')	(5.43')	(35.3')	(5.43')	(35.3')
3 SSE						
t1	.0716	.1253	.0650	.0977	.0639	.0640
t2	.0454	.0829	.0555	.0399	.0635	.0633
t3	.0811	.1266	.0614	.0636	.0638	.0635
t4	.0696	.1002	.0608	.0440	.0630	.0633
t5	.0658	.0928	.0597	.0688	.0635	.0634
t6	.0658	.0962	.0613	.0594	.0637	.0634
t7	.0771	.0971	.0552	.0622	.0633	.0635
t8	.0586	.0764	.0575	.0368	.0630	.0634
t9	.0629	.1005	.0540	.0419	.0633	.0633
t10	.0797	.1198	.0686	.0792	.0636	.0642
Average	.0678	.1005	.0599	.0594	.0635	.0635
Stand Dev.	.0102	.0168	.0043	.0185	.0003	.0003

Figure 4.35 Gap Closure Between Inlet/Outlet Shields and Primary Shield

for six simulations out of ten. The core impacts the primary shield at 35.3 feet above its base for two out of ten simulations. The core displaces about 1.5 inches and about 2.1 inches at 5.4 feet and 35.3 feet above its base. The primary shield displaces on the average of 0.7 inches at elevation 5.4 feet and 2.0 inches at elevation 35.3 ft.

As for the N-S direction, structural response spectra were calculated for the E-W direction, for damping ratios of 2 percent and 5 percent of critical. Figures 13 through 48 in Appendix F are examples of in-structure response of the core and inlet/outlet shield for two input motions and all three acceleration ranges.

At 1 SSE and 2 SSE, the primary shield response spectra have a large peak at roughly 2 Hz, which corresponds to the elastic natural frequency, and large spectral accelerations at high frequencies. The core response spectra have large peaks at frequencies ranging from 2 to 10 Hz and low spectral accelerations in the higher frequency range. The inlet/outlet shields have a large peak at approximately 7 Hz, and high spectral values for the higher frequencies. As described previously, this is caused by the sudden increase in stiffness due to the impact idealization, creating artificially high spectral accelerations at high frequencies for the two impacting structures. The core does not have high spectral accelerations for the higher frequencies because it does not impact the primary shield at these two levels of excitation. At 3 SSE, the response spectra have similar shapes except for the input time histories for which the core impacts the primary shield. Then the phenomenon described for the shields occurs for the core, although it is limited only to the top node, where impact occurs.

The analysis shows once more the sensitivity of the results to the input time history. The primary reasons are similar to those described previously. The results are also sensitive to the key action of the core, which was not the case in the N-S direction because the core was not keyed at its base in that direction.

Another important factor in the system response is the highly non-linear behavior of the primary shield. The relative displacement between the top and bottom nodes reached high values, even at 1 SSE.

Finally, the overall response was sensitive to the coupling of the inlet/outlet shields and the core. A study was conducted with input time history 1 at all three levels of excitation, without considering the process tubes. This analysis shows that the core displacements are larger for the case with the process tubes than those for the case without them. Therefore, it would seem that the inlet/outlet shields, pushed by the primary shield, pull the core.

4.4.5.4 Conclusions

In the N-S direction, both the core and the primary shield behave as sliding rigid bodies. At 1 SSE, the core displaces on the average of 0.8 inches and the primary shield between .007 inches and 0.2 inches. The core has about 20 percent chance of impacting the primary shield. The primary shield does not impact the 105-N structure. At 2 SSE, the core displaces on the average of 3.5 inches and the primary shield 2.0 inches. The core impacts the primary shield and the primary shield has about a 10 percent chance of impacting the 105-N structure. At 3 SSE, the core displaces on the average of 5.4 inches and the primary shield 4.0 inches. The core impacts the primary shield and the primary shield has about a 60 percent chance of impacting the 105-N structure.

In the E-W direction, the core and the primary shield exhibits large non-linear behavior and the inlet/outlet shields behave as sliding rigid bodies. Coupling of the core and the inlet/outlet shields through the process tubes have an important effect on the overall response as the inlet/outlet shields drive the core. At 1 SSE, the core displaces on the average of 0.5 inches at the bottom and 0.7 inches at the top. The primary shield displaces on the average of 0.02 inches at the bottom and 1.2 inches at the top. The inlet/outlet shield displace on the average of 0.7 inches. The inlet/outlet shield impacts the primary shield, at the top and has a 50 percent chance of impacting it at the bottom. Neither the core and the primary shield nor the 105-N structure and the primary shield impact each other. At 2 SSE, the core displaces on the average of 1.0 inches at the bottom and 1.6 inches at the top. The primary shield displaces on the average of 0.5 inches at the bottom and 1.9 inches at the top. The inlet/outlet shields displace on the average of 1.2 inches. The inlet/outlet shields impact the primary shield at both the top and bottom. The primary shield has a 40 percent chance of impacting the 105-N structure at the top. The core does not impact the primary shield. At 3 SSE, the core displaces on the average of 1.5 inches at the bottom and 2.1 inches at the top. The primary shield displaces on the average of 0.7 inches at the bottom and 2.0 inches at the top. The inlet/outlet shields displace on the average of 1.5 inches. The inlet/outlet shields impact the primary shield at both the top and bottom. The primary shield has a 60 percent chance of impacting the 105-N structure at the top. The core has a 20 percent chance of impacting the primary shield at the top.

4.5 Seismic Fragilities

In order to compute component and building failure probabilities, a probabilistic estimate for the acceleration level at which failure occurs must be available. Failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of loading. Loading may be described by peak ground acceleration, local spectral acceleration or moment, depending on the component and failure mode.

Building failure is defined to occur when significant cracking of concrete walls occurs so that anchorage capability is lost, or less likely, when actual collapse occurs. Building fragilities are derived in terms of peak ground acceleration. Component failure is taken as either loss of pressure boundary integrity or loss of operability. The component fragilities are related to the appropriate local response to permit an accurate assessment of the effects of common-cause seismic failures in the evaluation of the accident sequences.

Both site-specific fragility evaluations of structures and piping as well as site-specific and generic fragility evaluations of components were utilized. Aging effects were explicitly included only in the calculation of the core response (on which the process tube and core inlet piping fragilities were based) which utilized current material properties, and included consideration of failed keys in the core. Aging effects were not explicitly included in either the site-specific or generic evaluation of component fragilities. However, although very little data are available, an EPRI report (Ref. 29) has concluded that aging has very little effect on the seismic capacities of equipment.

4.5.1 Generic Fragilities

A generic data base of fragility functions for seismically-induced failures was developed in the SSMRP (Ref. 30). As a first step, all components were grouped into generic categories. For example, all motor operated valves located on piping with diameters between 2-1/2 and 10 inches were placed into a single generic category, and similarly, all motor control centers were placed into another generic category.

Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Subsystem Hardness Assurance Program. These data were critically examined for applicability and then statistically combined with the expert opinion survey data to produce the final fragility curves for the generic component categories as reported in Reference 30.

Subsequently, a review of 19 site-specific seismic PRAs was performed, with the results described in Appendix D. Finally, more recent site-specific component fragilities derivations contained in the Lawrence Livermore Laboratory data base (Ref. 31) were reviewed. Based on these reviews, contain of the SSMRP generic component fragility estimates of Reference 29 were updated.

The final generic categories and the corresponding fragility medians and uncertainties used for the N-Reactor seismic PRA are shown in Tables 4.18 and 4.19. These fragilities are used as the starting point in the simplified seismic PRA process. As in the use of any generic data base, one must be cognizant of the source of the data and the equipment to which it applies. An essential aspect of using this data is to perform a comprehensive plant walkdown and to examine the equipment in the plant being analyzed and compare it with the data base from which the generic fragilities were developed. Any deviation is noted and examined carefully, and new site-specific fragilities are developed as necessary (see Section 4.5.2).

Fragilities for electrical components represent a special problem in that there is a wide variety of electrical gear found within a plant. Typically, all this gear is enclosed in switchgear cabinets or motor control centers. The two lowest failure modes that were identified in the SSMRP fragility data base were relay chatter and inadvertent trip of circuit breakers. Virtually all the electrical switchgear and motor control centers in a nuclear power plant include these two types of components. Relay chatter is the weakest failure mode and, if included in a risk analysis without any circuit considerations, would be the dominant failure. Because, in many cases, circuits are protected by time delay circuits and because, in most cases, chatter of relays would not cause a change in the state of a system being controlled, the importance of relay chatter is greatly diminished. Other than for the reactor trip system, ECS actuation circuitry and the PCS control power circuitry, detailed electrical system models were not developed in the N Reactor internal events Level 1 PRA. (Operational records were used to develop failure frequencies for the electrical failure modes for the internal events analyses.) Lack of circuit models precluded inclusion of relay considerations in the seismic analysis. Hence, the N Reactor analyses did not to include relay chatter as a failure mode for electrical gear, but rather, included circuit breaker trip as the lowest functional failure mode. Note, however, restart plans for the N Reactor include an evaluation of this issue.

4.5.2 N-Reactor Site Specific Component Fragilities

During the plant walkdown, all components identified on the fault trees were examined. For most components, visual inspection showed that their capacity (overall anchorage and attachment configuration of controllers, lube oil reservoirs, etc.) was at least as good as those from which the

Table 4.18

Generic Component Categories

<u>Fragility Category</u>	<u>Component Class</u>	<u>Typical Components</u>	<u>Frequency (Hz)</u>
1	LOSP		ZPA
2	Relays	Ceramic Insulators	5-10
3	Circuit Breakers		5-10
4	Batteries		ZPA
5	Battery Racks		ZPA
6	Inverters		5-10
7	Transformers	4 kV to 480 V and 480 to 120 V	10
8	Motor Control Centers	Control for ESF Pumps and Valves	5-10
9	Aux. Relay Cabinets		5-10
10	Switchgear (Including Trans-formers, Buses and Breakers)	416 V and 480 V	5-10
11	Cable Trays		ZPA
12	Control Panels and Racks	RPS Process Control	5-10
13	Local Instruments	Misc. Pressure & Temperature Sensors	5-35
14	Diesel Generators	4160 ac Emergency Power Units	22
15	Horizontal Motors	Motor-Generator Sets	ZPA
16	Motor-Driven Pumps and Compressors	AFWS, RHR, SIS, Charging Pumps, Lube Oil Pumps, Diesel Starting Compressors	7
17	Large Vertical, Centrifugal Pumps (Motor-Drive)	Service Water Pumps	5
18	Large Motor-Operated Valves (<10")		ZPA
19	Small Motor-Operated Valves (>10")		ZPA
20	Large Pneumatic/Hydraulic Valves		ZPA
21	Large Check and Relief Valves		ZPA
22	Miscellaneous Small Valves (8")	Includes MSIV, ADP, and PORV	ZPA

Table 4.18

Generic Component Categories (Concluded)

<u>Fragility Category</u>	<u>Component Class</u>	<u>Typical Components</u>	<u>Frequency (Hz)</u>
23	Large Horizontal Vessels & Heat Exchangers	Pressurizer Relief Tank, CCW Heat Exchangers	ZPA
24	Small to Medium Heat Exchangers and Vessels	Boron Injection Tank	20
25	Large Vertical Storage Vessels with Formed Heads	RHR Heat Exchanger, Accumulator Tank	ZPA
26	Large Vertical Flat-Bottomed Storage Tanks	GST, RWST	
27	Air Handling Units	Containment Fan Coolers	5

Table 4.19
Generic Component Fragilities, in Units of Gravity (g)

Category	Generic Component	Median*	β_U	β_R
1	Ceramic Insulators	0.25	0.25	0.25
2	Relays	4.00	0.48	0.75
3	Circuit Breakers	7.63	0.48	0.74
4	Batteries	0.80	0.40	0.39
5	Battery Racks	2.29	0.31	0.39
6	Inverters	2.00	0.26	0.35
7	Dry Transformers	8.80	0.28	0.30
8	Motor Control Centers	7.63	0.48	0.74
9	Auxiliary Relay Cabinets	7.63	0.48	0.74
10	Switchgear	6.43	0.29	0.66
11	Cable Trays	2.23	0.34	0.19
12	Control Panels and Racks	11.50	0.48	0.74
13	Local Instruments	7.68	0.20	0.35
14	Diesel Generators	1.00	0.25	0.31
15	Horizontal Motors	12.10	0.27	0.31
16	Motor-Driven Pumps and Compressors	2.80	0.25	0.27
17	Large Vertical Centrifugal Pumps	2.21	0.22	0.32
18	Large Motor-Operated Valves (<10 inches)	6.50	0.26	0.60
19	Small Motor-Operated Valves (>10 inches)	4.83	0.26	0.35
20	Large Pneumatic/Hydraulic Valves	6.50	0.26	0.35
21	Large Relief, Manual, and Check Valves	8.90	0.20	0.35
22	Miscellaneous Small Valves	12.50	0.33	0.43
23	Large Horizontal Vessels and Heat Exchangers	3.0	0.30	0.53
24	Small to Medium Vessels and Heat Exchangers	1.84	0.25	0.45
25	Large Vertical Vessels With Formed Heads	1.46	0.20	0.35
26	Large Vertical Tanks With Flat Bottoms	0.45	0.25	0.35
27	Air Handling Units	6.90	0.27	0.61

R = Random Uncertainty.

U = Systematic Uncertainty.

*All medians in terms of spectral acceleration at 5% damping.

generic data base was derived. Thus, for these components, the generic data fragility estimates were used. A number of components were identified as not fitting in the generic fragility data base. These components were later examined for their importance on the system fault trees, and for those playing a role in the systems analysis, site-specific fragilities were derived. The resulting site-specific component fragilities are shown on Table 4.20.

It should be noted that the items identified had less margin than typical in the generic fragility data base. However, in each case, the median capacity of the component was well above that required for the SSE design basis earthquake.

4.5.3 N-Reactor Site Specific Building Fragilities

4.5.3.1 Method of Fragility Evaluation

The fragilities of the N-Reactor structures were generated using the basic methodology described in Reference 32, with certain modifications. The fragility of a structure can be expressed in terms of its peak ground acceleration capacity, A , as follows:

$$A = \hat{A}_m e_R e_U$$

In this formulation, $\hat{A}_m$ is the median peak ground acceleration capacity, and e_R and e_U are random variables with unit median, representing the inherent randomness about the median and the uncertainty in the median value. The variables e_R and e_U are assumed to be lognormally distributed with logarithmic standard deviations β_R and β_U , respectively. The properties of the lognormal distribution are presented in Reference 32.

For convenience, the median peak ground acceleration capacity, $\hat{A}_m$, was formulated as the product of the peak ground acceleration, A , and a median factor of safety against this ground motion level, $\hat{F}_m$.

$$\hat{A}_m = \hat{F}_m A$$

The median factor of safety, $\hat{F}_m$, was in turn expressed as the product of the following two median factors of safety:

- a. The median strength factor, $\hat{F}_s$, which is defined as the ratio of the median structure strength to the median structure loads for the ground motion input considered, and
- b. The median inelastic energy absorption factor, $\hat{F}_u$, which accounts for the ability of the structure to withstand seismic loads in excess of those corresponding to yield through ductile, nonlinear response.

Table 4.20

N Reactor Site Specific Fragilities

Component	Failure Mode	M_F	β_R	β_U	Response No.
13.8 KV A Bus	Sliding Failure	0.76g	0.25	0.30	1
13.8 KV B Bus	Sliding Failure	0.76g	0.25	0.30	1
4.16 KV A Bus	Sliding Failure	0.76g	0.25	0.30	1
4.16 KV B Bus	Sliding Failure	0.76g	0.25	0.30	1

The strength and inelastic energy absorption factors have associated logarithmic standard deviations, β_s and β_u . From the properties of the lognormal distribution, the logarithmic standard deviation associated with the total factor of safety is calculated as follows:

$$\beta = (\beta_s^2 + \beta_u^2)^{1/2}$$

These variabilities are composed of randomness and uncertainty, which are defined as follows:

- a. Randomness consists of variabilities that cannot be reduced by more detailed evaluation or data collection, and
- b. Uncertainty consists of variabilities resulting from lack of knowledge.

The only source of random variability reported in this section results from the effect of certain earthquake characteristics on the structure inelastic energy absorption capability. Uncertainties result from variables such as material strength, member capacity, member ductility, etc. Structure seismic response contributes additional variability to the structural fragilities. Logarithmic standard deviations for seismic response variability are not included in the values reported.

It should be noted that, in this study, structures are considered to fail functionally when inelastic deformations of the structure under seismic load are estimated to be sufficient to potentially interfere with the operability of safety-related equipment attached to the structure. The element and system ductility limits chosen for structures are estimated to correspond to the onset of significant structural damage. For many potential modes of failure, this is believed to represent a conservative bound on the level of inelastic structural deformation which might interfere with the operability of components housed within the structure. It is important to note that considerably greater margins of safety against structural collapse are believed to exist for these structures than many cases reported within this study. Thus, the structural element capacities reported herein should not be inferred as corresponding to structure collapse.

4.5.3.2 Development of Structural Capacities

The N-Reactor structural fragilities were expressed in terms of factors which account for structure ultimate strength and inelastic energy absorption capability. The basic techniques used to determine the median values and associated variabilities of these factors were essentially those described in Reference 32, with certain modifications.

4.5.3.2.1 Structure Element Ultimate Strengths

Two major considerations are involved in the determination of the ultimate strengths of individual structural elements. One is the definition of the strengths of the materials composing the members. The other is the determination of the ultimate strength capacities of the structural members given the type of loading, material strength, member configuration, etc.

N-Reactor material strength data were not researched. The median material strengths and variabilities listed in Table 4.21 were estimated based upon experience obtained from other nuclear plant fragility evaluations.

The median ultimate strength capacities of the structural elements were found using the median material strengths and member configurations (i.e., geometry, reinforcement, etc.) in conjunction with available predictive formulation or approaches. The approaches and formulations used were those appropriate for the type of element (i.e., shear wall, etc.) and loading (i.e., shear, flexure, etc.). They were typically found to provide essentially median centered capacities when compared to the results of available experimental testing. For example, the predictive equations used to determine the median ultimate strengths of the N-Reactor shear walls subjected to in-plane shear and flexure are presented in Reference 32.

Median strength factors, $\hat{F}_{sm}$, were calculated for individual structural elements as follows:

$$\hat{F}_{sm} = V_{um,i} / V_i$$

$V_{um,i}$ - Median ultimate strength for element i

V_i - Median load from response analysis for element i

Different median strength factors were typically calculated for each of the three ground motion levels since soil-structure interaction responses vary with input. The median strength factor for a structure was typically taken to be the lowest value of the individual elements composing its primary seismic load-resisting system. This is slightly conservative if the structural elements are ductile and redundant. In certain cases, load redistribution among such structural elements was considered when determining the structure strength factor.

Variability of the structural element ultimate strengths was considered to be composed of uncertainty since it is associated with a lack of knowledge. Uncertainty attributed to material strength was based upon the estimated variabilities listed on Table 4.21. Comparisons of the predicted strength capacities to the available test results provided

Table 4.21

Material Strength Properties

<u>Concrete Compressive Strength</u>			
Minimum Specified (psi)	Median (psi)	β	
3000	4900	0.17	
<u>Steel Reinforcement Yield Strength</u>			
Grade	Median (ksi)	β	
40	48	0.10	
<u>Structural Steel Yield Strength</u>			
ASTM Specification	Minimum Specified (ksi)	Median (ksi)	β
A7	33	41	0.10
A201 Grade B	32	40	0.10
<u>Concrete Block Compressive Strength</u>			
	Median (psi)	β	
	2000	0.20	

estimates of the uncertainty in the predictive strength formulations. Additional uncertainty attributable to variabilities associated with other sources, such as member geometry, reinforcement spacing, openings, workmanship, differences between field and laboratory conditions, accuracy of the predicted load distributions, etc., were also included.

4.5.3.2.2 Structure Inelastic Energy Absorption

Ductile structures have the ability to withstand seismic levels in excess of those corresponding to yield through nonlinear response. There are two basic methods for considering structure inelastic energy absorption. The first consists of time-history analysis with a mathematical representation of the structure that accounts for its nonlinear material characteristics. For example, yielding of the primary shield wall included in the core nonlinear analysis. In the second method, an inelastic energy absorption factor, F_u , is calculated as a function of structure ductility and ground motion characteristics. The ground motion necessary to cause a certain structure ductility level is the product of the inelastic energy absorption factor and the peak ground acceleration that initiates yielding. The following two approaches to estimate the inelastic energy absorption factor were used in this study:

- a. Riddell-Newmark response deamplification factor, and
- b. Effective frequency and damping.

With appropriate input parameters, these two methods provide comparable results. The former was developed based on a series of nonlinear analyses with ground motions producing broadbanded spectra as input. It was used in evaluating buildings 105-N and 109-N. The latter was used in evaluating building 181-N since it can more accurately account for ground motion with narrow-banded frequency content.

4.5.3.2.3 Riddell-Newmark Response Deamplification Factor

The Riddell-Newmark response deamplification factor (Ref. 33), ϕ_u , modifies the elastic ground response spectrum to create an equivalent reduced spectrum that accounts for structure nonlinearity. The median inelastic energy absorption factor, $\hat{F}_{um}$, corresponding to some ductility ratio, μ , is given by the following equation:

$$\hat{F}_{um} = \frac{S_{ae}}{S_{au}}$$

S_{ae} - Median spectral acceleration for median structure damping at the dominant structure frequency

S_{au} - Deamplified spectral acceleration at the dominant structure frequency

For frequencies in the amplified acceleration range (between about 2 Hz and 6 Hz for the median N-Reactor site ground spectrum):

$$S_{au} = \phi_u S_{ae} > S_{ahf}$$

$$\phi_u = (p\mu - q)^{-r}$$

$$p = q + 1$$

$$q = 3.0 \beta^{-0.30}$$

$$r = 0.48 \beta^{-0.08}$$

$$\beta = \text{Structure damping}$$

For frequencies greater than the frequency at which the median spectral acceleration returns to the peak ground acceleration (about 15 Hz) the deamplified spectral acceleration (denoted as S_{ahf}) is a function of and PGA as given below:

$$S_{au} = S_{ahf} = \mu^{-0.13} \text{ PGA}$$

PGA = Peak ground acceleration

The Riddell-Newmark response deamplification factor was based upon a series of nonlinear analyses utilizing single-degree-of-freedom (SDOF) fixed base models subject to time histories of large magnitude, long duration earthquakes. Nonlinear response of the N-Reactor structures would be expected to differ from the response calculated using these deamplification factors for the following reasons:

- a. The N-Reactor structures are founded on soil. As a result, soil-structure interaction effects may be significant,
- b. The N-Reactor structures are typically multi-degree-of-freedom (MDOF) systems, and
- c. Shear wall structures, such as the major N-Reactor structures, exhibit stiffness degradation and pinched hysteresis loops during nonlinear response.

To account for these differences, an effective ductility, μ_e , was used in the equations above.

The system ductility, μ_{sys} , for use with the Riddell-Newmark deamplification factor is a measure of the nonlinearity throughout the structure. For fixed-base SDOF structures, the system ductility is equal to the story drift ductility, μ_{st} . However, for MDOF structures, the system ductility may be less than the story ductility if the ratio

of the story demand to story capacity is not uniform through the structure. Also, nonlinear behavior has less effect on structures with significant SSI effects as compared to fixed base structures for the following reasons:

- a. Structure nonlinearity causes only slight frequency shift in system modes dominated by soil flexibility, and
- b. Increased damping due to hysteretic behavior is small compared to soil radiation damping.

The system ductility, μ_{sys} , can be related to the story ductility by a factor M.

$$\mu_{sys} = \left(\frac{\mu_{st} - 1}{M} \right) + 1$$

The median story ductility for typical nuclear plant shear walls is estimated to be about five. Buildings 105-N and 109-N essentially behave as rigid structures on flexible soil. This conclusion is based on the observation that there is little or no amplification of the foundation level input motion throughout the height of these structures. For these buildings, S_{au} was calculated using the equation for the rigid frequency range.

The Riddell-Newmark response deamplification factors were based only on large magnitude earthquakes. The nonlinear models did not include stiffness degradation and pinched hysteresis loops representative of shear walls. It is well known that lower magnitude earthquakes are not as damaging to structures and equipment as higher magnitude earthquakes with the same peak ground acceleration. The lower magnitude earthquakes have lower energy content and shorter durations which develop fewer strong response cycles. Structures are able to withstand larger deformations (i.e., higher ductility) for a few cycles compared to the larger number of cycles resulting from longer duration events. Also, stiffness degradation and pinched hysteresis imply less energy absorption capability for the same structure strength and ductility.

Earthquake magnitude effects and shear wall hysteretic behavior were accounted for by using an effective ductility, μ_e , in the Riddell-Newmark response deamplification factor approach. The effective ductility was calculated as follows:

$$\mu_e = 1.0 + C_D (\mu_{sys} - 1.0)$$

where the duration coefficient, C_D , is a function of the earthquake magnitude and μ_{sys} is the previously defined system ductility.

The results of the analyses performed in Reference 34 which included a nonlinear model of realistic shear wall behavior were used to provide estimates of the duration coefficient, CD, as a function of earthquake magnitude. In general, for earthquakes having magnitudes ranging from 4.5 to 6.0, a duration coefficient of 1.4 was determined to be appropriate by correlating the inelastic energy absorption factor from the Riddell-Newmark formulation to the results of Reference 34. Similarly, a duration coefficient of 0.7 was estimated for earthquake magnitudes in the 6.5 to 7.5 range. A duration coefficient of 0.7 was used for the N-Reactor structures to correspond to the hazard curve which was based on earthquakes above 5.5 mm.

4.3.3.2.4 Effective Frequency and Damping

This approach estimates nonlinear response of a single degree of freedom system with elastic frequency, f , and damping, β , for a given ground response spectrum using point estimates of an effective frequency, f_e' , and damping, β_e' . The effective frequency represents a value that is associated with some average nonlinear response cycle. The effective damping is the sum of the elastic damping and an effective hysteretic damping that is similarly associated with some average nonlinear response cycle.

Based upon Reference 34, the inelastic energy absorption factor corresponding to a certain ductility, μ , and ground motion can be formulated in terms of the effective frequency and damping as follows:

$$F_u = \mu (f_e'/f)^2 \frac{S_a(f, \beta)}{S_a(f_e', \beta_e')}$$

$S_a(f, \beta)$ - Spectral acceleration corresponding to the elastic frequency and damping

$S_a(f_e', \beta_e')$ - Spectral acceleration corresponding to the effective frequency and damping

From Reference 34, the effective frequency and damping can be approximated as follows:

$$f_e'/f = (1 - A) + A(f_s/f)$$

$$A = C_F(1 - f_s/f) < 0.85$$

$$\beta_e' = (f_s/f_e')^2(\beta + \beta_H)$$

$$\beta_H = C_N(1 - f_s/f)$$

f_s - Frequency based upon secant stiffness at ductility level analyzed

Reference 34 determined values of coefficients C_F and C_N to provide best fits between this simplified approach and results of nonlinear analyses using actual recorded earthquake ground motions and a nonlinear model that accounted for pinched shear wall behavior. These coefficients were found to vary with earthquake magnitude and number of strong motion cycles. Assuming that the N-Reactor seismic risk results primarily from earthquakes with strong motion duration greater than 15 seconds, values of C_F and C_N were estimated to be 2.7 and 0.11 from Table 4-2 of Reference 34.

4.5.3.3 Structure Fragilities

Fragilities for the N-Reactor structures are listed in Table 4.22. In general, several potential failure modes were investigated for each structure. Fragilities for only the governing failure modes are reported. The failure modes for each building are described below.

4.5.3.3.1 Building 105-N and Building 109-N

The part of building 105-N evaluated in this study is designated as Area 2 located at the center of building 105-N complex and is separated from the rest of the building on three sides. The separation from the rest of the building is accomplished by providing a continuous 2 to 3 inches wide expansion joint. This expansion joint extends vertically for the full height of the building and begins at the top of the foundation. On the fourth (south) side, it is contiguous with the 109-N heat exchanger building. This central portion of building 105-N encloses the process area, inlet/outlet piping areas, inner/outer rod rooms and other safety-related equipment. In the rest of this report, Area 2 of this building shall be referred as "building 105-N."

Building 105-N is constructed primarily of reinforced concrete. The principal structural walls are from 3 feet to 5 feet 9 inches thick and provide nuclear shielding as well as confinement capability. Both regular and high density concrete were used in the construction. The Area 2 of building 105-N is approximately 150 feet long (east-west direction), 130 feet wide (north-south direction) and 82 feet high. It shares a 5-foot thick reinforced concrete wall with Building 109-N on the south side. Thus, buildings 105-N and 109-N are coupled in the east-west direction by this common wall. In the north-south direction, the two buildings are tied together by dowels and eight additional clusters of Grade 40 No. 18S reinforcing bars. The primary reactor shield is supported by building 105-N, but is otherwise structurally independent.

The primary lateral force resisting system of building 105-N consists of major concrete floor slabs acting as diaphragms and the concrete shear walls. The controlling failure mode of building 105-N was found to be in-plane flexural failure of the major north-south shear walls located

Table 4.22
Structural Fragilities

<u>Structure</u>	<u>A_m</u>	<u>beta-r</u>	<u>beta-u</u>	<u>Consequence</u>
Building 105	1.0g	0.03	0.23	Damage to equipment throughout the structure.
Building 109	1.7g	0.03	0.23	Damage to equipment throughout the structure.
Building 109-NT	0.35g	0.147	0.299	Fails G,A,C systems
Core Inlet Piping	0.72g	0.18	0.33	Damage to process tubes
Primary Shield	0.90g	0.03	0.30	Damage to process tubes.
Building 181-N	0.47g	0.08	0.20	Damage to components housed within the superstructure above the operating floor.
Building 182-N	1.025	0.147	0.273	Fails G,A,C systems.
Concrete Silo	0.54g	0.06	0.21	Failure of piping attached to the silo and routed into the ground.
Demineralized Water Storage Tank	>1.25g			

Notes:

1. A_m is the median free-field peak ground acceleration capacity.
2. B_R and B_U are the lognormal standard deviations associated with randomness and uncertainty on structure strength and inelastic energy absorption only. Seismic response variability is not included.

about 70 feet on either side of the centerline of building 105-N. There are fewer concrete shear walls in the north-south direction to resist lateral force. The median PGA capacities of this mode of failure was determined to be 1.0g, 1.1g, and 1.2g for the one, two, and three times SSE level of input ground motion, respectively. This failure mode corresponds to gross structural failure and damage to the equipment contained in this building. This capacity considered the soil replacement effect developed in Reference 13 for the one times SSE level. This soil replacement factor accounts for the difference in the structural response between the cases in which the control point is defined at the foundation level and at the ground surface. The same soil replacement factors were assumed for the capacities for the two and three times SSE ground motion input.

Building 109-N houses most of the RCS piping, the pressurizer, steam generators and RCS pumps. The steam generators and RCS pumps are located in six isolated cells and are serviced from a large pipe gallery located on the north side of the cells. The seventh cell which is located at the center of building 109-N contains auxiliary equipment. Each of the six steam generator cells and the auxiliary cell is separated from adjacent cells and from the pipe gallery by shielding walls 5 ft thick.

Building 109-N is approximately 390 feet long (east-west direction), 56 feet high and 145 feet wide (north-south direction). Shielding walls are 5 feet thick, except for the upper section of the south wall which is 5 feet-9 inches thick. The 3 feet-6 inches thick reinforced concrete roof is located about 40 feet above ground level. The primary lateral force resisting system consists of concrete slabs and shear walls. At the roof level, clusters of reinforcing steel were provided to tie buildings 105-N and 109-N together in the north-south direction.

The governing failure mode of this building was found to be the in-plane flexural failure of the auxiliary cell shielding wall in the north-south direction. The median PGA capacities of this failure mode at the input levels of one, two, and three times SSE were determined to be 1.7g, 1.9g, and 2.0g, respectively. The potential consequence of this failure mode would be damage to the safety-related equipment housed in this structure. Failure of the east-west oriented walls was evaluated and was found not to govern.

Other structural components of buildings 105-N and 109-N which were evaluated included: the common wall and the north-south structural ties between the two buildings. The common wall is 5 feet thick and 390 feet long. The middle portion of this wall is an integral part of Building 105-N and the rest is an integral part of building 109-N. This wall was designed to act as a unit by providing shear keys and steel dowels at the joints between the 105-N part and the 109-N part of the wall. The shear keys and dowels were provided throughout the wall height. The median PGA capacity of this common wall was determined to be greater

than 1.6g. The north-south structural ties between buildings 105-N and 109-N are provided by a total of eight clusters of No. 18S reinforcing bars located at elevation 40 feet and elevation 60 feet with a total area of 848 square inches. The median PGA capacity of this component was also determined to be greater than 1.6g.

4.5.3.3.2 Primary Reactor Shield

The primary reactor shield is a box-shaped structure constructed of reinforced concrete and structural steel that encloses the core. The shield has a total height of 42.5 feet with inside plan dimensions of 34 feet-2 inches and 41 feet-1 in. The top shield has a total thickness of about 5 feet-5 in. It is supported by built-up structural steel girders that span between the left and right shields. The concrete walls of the primary shield are enclosed by structural steel plate that was included to act as forms for the heavyweight concrete. The left and right shields, which are located on the west and east sides of the primary shield, are 3 feet - 7 in. thick. Core cooling tubes and horizontal control rods penetrate the left and right shields. Bearing plates separate the primary shield from the core foundation. Steel sections are welded to the bearing plates beneath the left and right shields and grouted into the foundation. Vertical reinforcement is not doweled across the bearing plates. The east and west sides of the top shield are separated from 3 feet thick floor slabs of building 105-N by 2 inches wide gaps.

The inlet and outlet shields, which are penetrated by the process tubes, are built into the front and back faces of the primary shield. The inlet and outlet shields bear on roller assemblies and are separated from the primary shield at the top and sides by 0.75-in.-wide gaps. Out-of-plane displacements of the inlet and outlet shields are restrained by the primary shield. They are free to expand or contract under thermal strains in-plane.

Under E-W seismic loads, the left, right, and top shields essentially perform as moment frames. The front and back faces of the primary shield that enclose the inlet and outlet shields function in the same manner. Because reinforcement is not doweled to the foundation, these frames are essentially pin-supported at the base. Moment capacities of the various structural components considered the ultimate strength of the reinforced concrete sections plus additional contribution from the structural steel plate. Preliminary analyses indicated that hinging occurs at relatively low ground motion levels. Consequently, nonlinear properties of the primary shield were included in the core analysis.

Results of the E-W core response analysis show that impact between the primary shield and the inlet and outlet shields occurs even at the one times SSE input level. Impact between the primary shield and the building 105-N floor slabs occurs at the two times SSE level. Gross structural failure due to impact in the E-W direction is considered

unlikely. The impact loads are of very short duration and thus have only limited energy content. Overall structure failure is less likely for input with smaller energy content. Past earthquake performance of conventional structures that are less massive than the primary shield shows that, while there are instances of gross structural failure, impact usually only causes local concrete spalling. Similarly, impact would be expected to cause concrete crushing and failure of structural steel welds local to the zone of impact. Even if some form of structural failure were to occur, translation of the primary shield would still be limited by the adjacent building 105-N floor slabs and thus retain overall structure integrity.

Structure impact can cause motions that may affect components that are sensitive to high frequency input. Such motions are included in the primary and inlet/outlet shield spectra. Impact may also damage equipment anchorages close to the zone of impact. The process and cooling tubes are located away from the impact regions and would probably not be damaged by local impact effects. Also, they are supported by penetration sleeves embedded in the primary and inlet/outlet shields. Loss of support for the process and cooling tubes is thus less likely. The only critical equipment in the vicinity is the ball hoppers, the control rods, and sensors.

Even though steel sections are welded to the base plates, their placement is such that resistance against horizontal shear in the N-S direction will be relatively low. In the N-S core analysis, the primary shield was permitted to slide when its static frictional resistance is overcome. N-S sliding displacements are limited to a maximum of 5 inches by the raised concrete pedestal supporting the core.

The 35 inches wide portions of the primary shield below the inlet and outlet shields are composed of the structural steel plate filled with unreinforced concrete. These areas are probably more vulnerable to impact-induced structural failure. Impact with the raised concrete pedestal may shear these sections off at the corners. At the one, two, and three times SSE input levels, impact occurs for none, one, and six of the ten time-histories, respectively. If impact is conservatively assumed to correspond to structure failure, this would imply that the median capacity is just below the three times SSE level and a lower bound capacity is between the one and two times SSE levels. Using statistical methods, a fragility for impact based only on the time-history data was estimated. This fragility is somewhat conservative for the following two reasons:

- a. The coefficient of friction used in the analysis was reduced from the static value assuming that the peak vertical acceleration acts concurrently throughout the duration of sliding, and
- b. The actual failure displacement is some value slightly greater than the separation gap.

Eliminating these two conservatisms, a median peak ground acceleration capacity of 0.90g was estimated. Failure of the primary shield due to N-S impact may damage the process tubes.

4.5.3.3.3 Building 181-N

The 181-N river pumphouse houses components of the N-Reactor emergency core cooling system. Building 181-N consists of a substructure and a superstructure. The substructure includes all portions at and below the operating floor at elevation 421 feet. It is of heavy reinforced concrete construction. The building is on a mat foundation that is 5 feet thick and 94 feet by 124 feet in plan. The top of mat is at elevation 360 feet. The operating floor varies from 1 to 3 feet in thickness. Most of the major equipment components are mounted to the operating floor. Several shear walls ranging in thickness from 3 to 4 feet stand on the mat foundation and support the operating floor. Grade on the east side is located at elevation 421 feet. Backfill slopes down to the top of the mat foundation on the west side.

The superstructure rises above the operating floor. It encloses the emergency core cooling system pumps and diesel engines. The superstructure is 37 feet by 51 feet in plan and about 18 feet high. The concrete roof slab varies in thickness between 6 and 8 inches. The superstructure has twelve reinforced concrete columns with interconnecting beams to form moment frames. Twelve-in.-thick reinforced concrete block (masonry) wall panels between the exterior columns enclose the superstructure. These panels have vertical steel reinforcement and horizontal joint reinforcement. They are mortared into the concrete columns and doweled to the operating floor. Masonry bond beams at the tops of the panels are reinforced and doweled to the columns.

The gantry crane travels in the E-W direction on rails mounted to the operating floor. The main hoist has a 60 ton capacity. The span between the tracks is 104 feet and the distance between end stops is about 120 feet.

Reference 11 describes a deterministic seismic and tornado evaluation of building 181-N performed to assess conformance to conservative acceptance criteria. Seismic analysis was performed with the same overall soil-structure interaction model used in this study to generate probabilistic responses. Applied loads were compared to available capacities for a number of structural components. This evaluation found that seismic capacity of building 181-N is limited by the in-plane moment capacity of the N-S block walls at the west side of the superstructure.

Fragility evaluation of building 181-N focussed on the seismic capacity of the superstructure in the N-S direction. Major features of this evaluation included the following:

- a. Block wall panel ultimate strength and ductility,
- b. Load redistribution, and
- c. Inelastic energy absorption.

Design of reinforced masonry has traditionally been based upon working stress principles with material stresses limited to values well below yield. However, testing reported in References 35 to 38 demonstrate the following:

- a. In-plane moment capacities of reinforced masonry walls can be determined using the ultimate strength design principles for reinforced concrete, and
- b. Under-reinforced masonry walls are ductile and can withstand deformations in excess of yield.

Accordingly, the in-plane flexural strengths of the superstructure block walls were determined following the ultimate strength provisions for reinforced concrete shear walls described in Reference 39. The following considerations were included:

- a. Appropriate median concrete block and reinforcing steel properties
- b. Code strength reduction factor excluded
- c. Slight reduction in available in-plane capacity due to concurrent out-of-plane seismic loads
- d. Appropriate modifications for door openings

The block wall panels and concrete moment frames on the west side of the superstructure attract greater seismic load because the columns at the louver openings are shorter, and thus stiffer. Yielding will initiate in the west block wall panels. However, because these panels are under-reinforced, they are able to withstand deformations in excess of yield through ductile behavior. After the west panels yield, additional superstructure seismic inertial loads will redistribute to the east panels.

To account for load redistribution, the overall N-S superstructure seismic capacity was expressed in terms of the load-displacement relationship for an equivalent single-degree-of-freedom system. This is appropriate since the roof slab is sufficiently stiff to constrain displacements of the entire superstructure. The overall load-displacement relationship was developed in terms of the individual relationships for the west and east sides, including progressive

yielding of various structural components under increasing load.

Failure of the superstructure was defined to occur when the ductility limit of the block wall panels was achieved. Testing on concrete block panels failing in flexure shows that ductilities ranging from about three to five and story drifts of about 1 percent can be achieved. A median ductility of four was estimated for the superstructure panels.

Including load redistribution resulted in the following:

- a. Increase in the effective structure yield capacity by about 10 percent,
- b. An overall system ductility of about 3, and
- c. A roof displacement at failure of about 0.85 inches (0.4 percent drift).

The inelastic energy absorption factor for this failure mode was determined using point estimates of the effective frequency and damping. Because responses of the superstructure and substructure are essentially decoupled, it is possible to treat the superstructure as a single degree of freedom system subjected to seismic input defined by the operating floor spectrum. The effective frequency and damping approach is better suited to determine superstructure nonlinear response because the operating floor spectra are narrow-banded in the frequency range of interest.

An effective frequency of about 4.2 Hz and effective damping of 12 percent were estimated. The frequency shift includes the effect of nonlinear response as well as elastic stiffness reduction due to cracking. Seismic input to the superstructure consisted of the median operating floor spectra. Median inelastic energy absorption factors of 1.38, 1.22, and 1.08 were estimated for the one, two, and three times SSE level input. These values are less than those that would be obtained from the Riddell-Newmark approach with a system ductility of three because frequency reduction shifts superstructure response into a region of higher input.

Median peak ground acceleration capacities of 0.47g, 0.47g, and 0.41g were estimated for the one, two, and three times SSE level. Failure of the superstructure is expected to result in damage to the ECCS diesel-driven pumps and their associated peripherals and the third diesel which is located just outside the superstructure on the operating floor.

4.5.3.3.4 Concrete Silo Tank

The concrete silo tank is part of the emergency core cooling system bypass. It is a vertical cylinder with an open top that is constructed of reinforced concrete. Outside and inside diameters of the silo are 13

feet and 11 feet, respectively. The top of the silo is 49 feet above grade. It is supported by a mat foundation that varies in plan to a maximum of 26 feet in diameter. The bottom of the foundation is about 14 feet below grade. Various large-diameter pipes are attached to the inside and outside. The silo contains water with the overflow line located 8 feet below the top.

Reference 40 describes a recent analysis of the silo to determine its conformance to Hanford design criteria. Seismic response was based upon a finite element model of the silo and attached piping. Elastic half-space soil stiffnesses were included. This evaluation concluded that the silo has sufficient capacity against seismic loads and overturning about its base. Certain overstressed pipe supports were identified.

Detailed SSI analysis to determine probabilistic responses was not performed for the silo. Instead, approximate elastic and nonlinear analyses were conducted to estimate seismic response before and after the initiation of uplift. The elastic analysis was based on a lumped mass model with the silo represented by a series of beam elements. Approximate soil-structure interaction analysis accounted for strain-dependent soil properties, foundation embedment, soil impedance frequency dependence, and soil radiation damping. Based on the elastic analysis, base uplift due to overturning is considered more likely than loss of structural integrity.

Initiation of base uplift does not necessarily correspond to a loss of stability. At higher ground motion levels, the silo will rock about its base as a rigid body. Resistance against overturning is provided by the dead weight restoring force which is eccentric about the pivot axis and lateral earth pressure. The reserve energy technique (Ref. 41) provides a tractable method of accounting for nonlinear response of a rigid body rocking system. Reference 42 found that this method adequately estimates the collapse capacities of unreinforced concrete block walls in comparison to results from nonlinear time-history analyses.

The reserve energy method was used to estimate nonlinear response of the silo after base uplift. Attached piping was judged to fail at a lateral displacement of about 2 inches. A median peak ground acceleration capacity of 0.54g was estimated for failure of attached piping due to silo rocking.

4.5.3.3.5 Demineralized Water Storage Tank

The demineralized water storage tank (DWST) is a ground-mounted vertical tank. It has a diameter of 75 feet and shell height of 32 feet and is fabricated of A-283 Grade C steel. The shell thickness varies from 0.25 inches to 0.5 inches. The bottom plate is 0.25 inches thick. The overflow line is located about 30 feet above the tank base.

The DWST was originally unanchored to its ring wall foundation. A recent seismic evaluation (Ref. 43) was performed to determine conformance to Hanford design criteria. This evaluation found that the tank will slide at the SSE and that attached piping have insufficient flexibility to accommodate the sliding displacement. To increase the tank seismic capacity, reinforced concrete buttresses were installed around the perimeter. These buttresses were anchored to the tank and its foundation.

The DWST is considered to have substantial seismic capacity by inspection. Rather than perform a detailed fragility evaluation, a conservative bounding analysis was conducted to confirm this judgement. This bounding analysis used the following conservative assumptions:

- a. Potential soil-structure interaction effects were neglected,
- b. Tank-fluid impulsive mode response was based upon the peak spectral acceleration for the 5 percent damped median ground spectrum, and
- c. The DWST was assumed filled to the overflow line rather than the normal operating level.
- d. Conservative, rather than median, allowable stresses and capacities were used.

Besides restraining the tank against sliding, the buttresses also resist uplift because they cover the portion of the bottom plate that extends past the shell. This was considered in the fragility evaluation. The following stresses were compared to allowable values:

- a. Shell meridional compressive stress
- b. Shear stress in fillet weld between the shell and bottom plate
- c. Bottom plate shear stress
- d. Bottom plate compressive stress
- e. Concrete bearing stress at bottom plate interface
- f. Buttress reinforcement tension stress
- g. Shear transfer between buttress and ring wall

Based upon this evaluation, it was concluded that the DWST has a median peak ground acceleration capacity greater than 1.25g.

4.5.3.3.6 Emergency Raw Water Tank

The fragility analysis for the Emergency Raw Water (ERW) tank was performed by Westinghouse-Hanford Co. in support of the seismic PRA as given in Reference 44. This section is taken directly from Reference 44. Several failure modes of the ERW tank were analyzed. They are failures due to sliding and uplifting of the tank, and tensile failure or buckling of tank shell. The seismic capacity of the ERW Tank was derived for each failure mode using appropriate methodologies.

The original tank simply rested on the ring foundation without using any physical connections such as anchor bolts. Buttresses were installed, the purpose being to provide a physical connection between the tank and the ring foundation and to resist tank sliding during an earthquake event. It was assumed that if the connections were acceptable the tank would not fail. This is true because the ring foundation and the soil surrounding by the ring foundation provides sufficient resisting force against tank sliding.

For the sliding mode, the seismic capacity of the buttresses is obtained considering three different failures: compressive failure, tension failure and shear failure. The weakest link of the buttress in compressive failure is the "tie-ins" which are grouted into the ERW Tank existing ring foundation at one end and embedded into the newly erected buttress at the other end. Due to limited length of "tie-in", the development length is much shorter than the ACI Code requirement. Thus, during the compressive and tensile loading the tie-ins can be pulled out. However, different moment arm lengths give different tensile and compressive strengths of the buttress for the sliding failure mode. The buttress failure in tension and shear is governed by the strength of the straps which are welded to the bottom plate of the ERW tank at one end and embedded in the reinforced concrete of buttresses at the other end. The size of the straps governs the tensile and shear strengths of the buttresses. These strengths are obtained and used in the development of the total seismic capacity of the ERW Tank in the sliding mode. The seismic loading of the sliding mode is derived by considering deadweight, hydrostatic and hydrodynamic loadings of both water and the ERW tank.

It is judged that the pure uplifting would not happen due to the extremely heavy water content and low aspect ratio of height versus diameter of the tank. However, the tank and its foundation support may overturn if the soil bearing stress due to dead load and seismic loading is greater than the soil bearing allowable stress.

For the tensile failure mode of the tank shell, the tensile stress is obtained using the overturning moment divided by the section modulus of the tank. Hydrostatic pressure contributes some tensile stress in the tank shell wall. The tank fails in tensile stress if the overall tensile stress exceeds the yield strength of the material. For the buckling failure mode, the highest compressive stress obtained from finite element analyses is used to compare with the calculated buckling load capacity.

Details of the analysis of each of these modes are given in Reference 44. The fragilities of these modes in terms of the median peak ground acceleration and uncertainty parameters are summarized below.

<u>Failure Mode</u>	<u>Median (g)</u>	<u>beta-r</u>	<u>beta-u</u>
Sliding			
No Friction	1.55	0.20	0.50
With Friction	1.65	0.20	0.50
Uplifting	1.09	0.25	0.20
Tensile	8.55	0.34	0.35
Buckling	0.56	0.34	0.35

Clearly, the tank wall buckling mode dominates the capacity of the ERW tank in its current configuration.

4.5.3.3.7 Building 182N

A seismic fragility analysis was performed by Westinghouse-Hanford personnel for the 182N building in support of the N Reactor probabilistic risk assessment as reported in Reference 12. This section is taken directly from Reference 12. The analysis is based on safe shutdown earthquake (SSE) design analyses recently completed for the buildings using the FLUSH soil-structure interaction program (Ref. 19). These analyses were reviewed for main member identification and loads, and building fragilities were then developed.

Seismic building fragilities were estimated using the recent SSE analyses and the methodology outlined in Reference 30. A (total) fragility factor F relates the median ground acceleration causing failure to the 0.25g SSE value. Beta-r and beta-u are the lognormal standard deviations for randomness and uncertainty, respectively. The fragility factor is the product of factors reflecting the following effects:

SSE	SSE analysis margin
S	Strength
mu	Ductility
SA	Ground motion and associated response spectra
del	Damping, actual vs design
M	Modeling assumptions
MC	Modal combination
SSI	Soil-structure interaction

Except as remarked in the tables, the fragility and uncertainty factors are taken from Table 2 of Reference 30.

Figure 4.36 shows two sections through building 182-N, and indicates the column lines which will be referred to subsequently. Table 4.23 summarizes the main load carrying members and their loads at the design (1 SSE) level.

The bolt double shear condition for the vertical bracing is considered controlling, because of the low system ductility relating to the dimension of the connection compared to the bracing dimensions. The factors of safety for this failure mode are shown on Table 4.24.

The other condition considered is the horizontal shear in the ground floor slab on the south side of the building which has several openings between the floor and wall. These results only apply to N-S excitation. The factors of safety for this failure mode are shown on Table 4.25

The column anchor margin of safety is considered not to be controlling since the net strength and ductility is expected to be near that of the slab factor, and the vertical bracing precedes it. Buckling of the vertical bracing is not considered significant since there is no steady compression load and each bay is doubly braced. The brace connection fragility of Table 4.24 controls and is taken to define the building fragility.

Based on the overall factor of safety of 4.1 as shown in Table 4.24, the median failure level for the building 182-N is taken as $(4.1) \times 0.25g = 1.025g$. The random and modeling uncertainties are as shown on Table 4.24.

4.5.3.3.8 Building 109-NT

A seismic fragility analysis was performed by Westinghouse-Hanford personnel for the 109-N turbine extension building in support of the N-Reactor probabilistic risk assessment as reported in Reference 12. This section is taken directly from Reference 12. The analysis is based on safe shutdown earthquake (SSE) design analyses recently completed for the buildings using the FLUSH soil-structure interaction program (Ref. 19). These analyses were reviewed for main member identification and loads, and building fragilities were then developed.

Seismic building fragilities were estimated using the recent SSE analyses and the methodology outlined in Reference 30. A (total) fragility factor F relates the median ground acceleration causing failure to the 0.25g SSE value. β_r and β_u are the lognormal standard deviations for randomness and uncertainty, respectively. The fragility factor is the product of factors reflecting the following effects:

Figure 4.36 Building 182N Elevation Sections

Table 4.23 182N Main Member Identification

<u>Member</u>	<u>Location</u>	<u>MS¹</u>	<u>Load</u>	<u>Mode</u>
Vertical bracing	Grd-roof, NS	0.10	74k	Compression (2)
	NS	0.49	74k	Short buckling (3)
	EW	0.20	82k	Bolt, double shear
Column anchor	Grd at brace	0.02	95k	Shear
Block walls	Grd-roof	0.84	870 ft-lb	Bending, out-plane
Floor slab, r/c	Grd, S wall	0.10	305k	Shear, longitudinal
Wall, r/c	Bsmt-Grd	0.75	1200k	Shear, longitudinal

Remarks:

- (1) Margin of safety MS = capacity/load - 1.0
- (2) No +33% credit for seismic taken in AISC eq (1.6-1b) for for vertical bracing membrane+bending compression.
- (3) Doubled vertical bracing modification was made.

Table 4.24 182N Bracing Connection Fragility

<u>Effect</u>	<u>F</u>	<u>Beta-r</u>	<u>Beta-u</u>	
SSE	1.2			
S	1.8	0.09	0.15	Ultimate/yield ratio
mu	1.1	0.11	0.22	Connection role in system ductility
SA	1.3	0.19	0.10	Broad nature of El Centro t/h
del	1.1	0.07	0.07	
M	1.2	0.0	0.15	Inertial load method in SSE analysis
MC	1.0	0.15	0.0	t/h analysis
<u>SSI</u>	<u>1.0</u>	<u>0.04</u>	<u>0.17</u>	FLUSH analysis
Total	4.1	0.29	0.37	

Table 4.25 182N Slab-wall Connection Fragility

<u>Effect</u>	<u>F</u>	<u>Beta-r</u>	<u>Beta-u</u>
SSE	1.1	-	-
S	3.8	0.12	0.18
mu	1.2	0.05	0.10
SA	1.3	0.19	0.10
del	1.1	0.07	0.07
M	1.2	0.0	0.15
MC	1.0	0.15	0.0
SSI	1.0	0.04	0.17
Total	7.8	0.28	0.33

SSE	SSE analysis margin
S	Strength
mu	Ductility
SA	Ground motion and associated response spectra
del	Damping, actual vs design
M	Modeling assumptions
MC	Modal combination
SSI	Soil-structure interaction

Except as remarked in the tables, the fragility and uncertainty factors are taken from Table 2 of Reference 30.

Figures 4.37 and 4.38 show two sections through building 109-NT, and indicate the column lines which will be referred to subsequently. Table 4.26 summarizes the main load carrying members and their loads at the design (1 SSE) level.

The single diagonal D-6 supports the mezzanine level laterally and shows a negative margin against the AISC short column buckling equation. The seismic loading is considered sufficiently deformation controlled to warrant a high fragility factor. The factors of safety are shown on Table 4.27. Similar remarks apply to the double EW bracings. The bolt shear at the D-6 connections is considered critical, because a relatively low system ductility is expected.

The NS and EW load results in the substructure are shown separately for two reasons. First, the analysis considered separate sets of column rebar for the two directions. Second, no credit was taken for the roof, mezzanine or floor slab abilities to transfer EW loadings toward the stiff substructure shear wall at column line V. Assuming all the calculated EW column T and R loads from the floor up are transferred through the slab, a 100 psi shear stress is calculated between columns T and V, while the allowable is 160 psi. Similarly, a 70 psi shear stress is found in the worst case between columns T and R. The transfer mechanism is in-plane shear which is much stiffer than column bending. Thus, it is considered likely that the column T and R superstructure loads will be transferred to the shear wall, and the EW bending moments in the substructure columns are largely discounted.

NS loading on the substructure is through a frame with columns at V, T and R. The SSE analysis checked demand versus allowable for each column independently. Since no unacceptable results were obtained for the connecting beams, a limit margin is obtained by adding demands as well as capacities and comparing the two. This result is shown as a sum of columns effect in Table 4.28.

The bolt shear condition is expected to be similar to the 182-N result, except that the D-6 N-S diagonal has an SSE margin of -0.49 if no upgrade is made. If the upgrade is made, the margin is assumed to be 0.20, the same as in the 182-N case.

Figure 4.37 Building 109NT Plan and NS Elevation

Table 4.26 109NT Main Member Identification

<u>Member</u>	<u>Location</u>	<u>MS¹</u>	<u>Load</u>	<u>Mode</u>
NS Diagonal D-6	Grd-mezz	-0.22	25k	Short buckling
		-0.49		Bolt shear
NS Diagonal D-S	Mezz-roof	-0.29	92k	Bolt shear
EW Diagonal D-11	Grd-mezz, col V	-0.60	69k	Buckling
		0.92		Connection
EW Diagonal D-14	Grd-mezz, col T	-0.05	52k	Buckling
		0.90		Connection
EW Diagonal D-13	Mezz-roof, col T	-0.54	88k	Buckling
		0.50		Connection
EW Diagonal D-12	Roof-pltf, col T	-0.19	42k	Buckling
		0.17		Connection
Column anchor	Grd, col V-54	-0.13	245k	Bolt shear
Column, R/C	Sub, col V	-0.06	1800 ft-k	Bending, NS
Column, R/C	Sub, col T	-0.31	2600 ft-k	Bending, NS
Column, R/C	Sub, col R	-0.16	1900 ft-k	Bending, NS
Sum columns, R/C	Sub, col V-T+R	-0.19	6300 ft-k	Bending, NS
Column, R/C EWT1	Sub, col T	-0.58	2400 ft-k	Bending, EW
Column, R/C EWRS	Sub, col R	-0.29	1400 ft-k	Bending, EW

Remarks:

1. Margin of Safety MS = capacity/load - 1.0

Table 4.27 109NT D-6 Connection Fragility

<u>Effect</u>	<u>F</u>	<u>Beta-r</u>	<u>Beta-u</u>	
SSE	0.51			No fix. Use 1.2 with proposed fix
S	1.8	0.09	0.15	Ultimate/yield ratio
mu	1.1	0.11	0.22	Connection role in system ductility
SA	1.3	0.19	0.10	Broad nature of El Centro t/h
del	1.1	0.07	0.07	
M	1.0	0.0	0.20	Torsion, soil, inertial loads
MC	1.0	0.15	0.0	t/h analysis
<u>SSI</u>	<u>1.0</u>	<u>0.04</u>	<u>0.17</u>	FLUSH analysis
Total	1.4	0.29	0.39	

Table 4.28 109NT Basement Column Fragility

<u>Effect</u>	<u>F</u>	<u>Beta-r</u>	<u>Beta-u</u>
SSE	0.81		
S	2.3	0.12	0.18
mu	1.7	0.11	0.15
SA	1.3	0.19	0.10
del	1.1	0.07	0.07
M	1.5	0.0	0.20
MC	1.0	0.15	0.0
<u>SSI</u>	<u>1.0</u>	<u>0.04</u>	<u>0.17</u>
Total	6.8	0.30	0.37

Three modeling effects have been identified:

(a) Torsion was not accounted for in the two dimensional analyses. Torsion is associated with EW excitation in that the superstructure mass center is about 24' north of the basement shear wall. This is expected to produce NS response in the building. A rough estimate places the contribution at 20%. Thus $F = 0.83$.

(b) Soil elements outside the basement shear wall in the FLUSH runs were not connected to the wall. The soil could be stiffer than the basement structure and therefore raise the 1.4 Hz fundamental mode basically tied to basement deformation. If the ground floor is fixed, a 3 Hz mode results which places the superstructure response at the peak of the design acceleration response spectrum, about 50 times higher than the 1.4 Hz response. Two observations lead to a soil stiffening effect -- deeper soil columns typically have 2-3 Hz modes, and a viscous boundary (FLUSH manual, Appendix C) estimate giving a damping fraction of 2-3 (overdamped). Thus, $F = 0.75$, as stiffening is not expected to be complete.

(c) The SSE analysis procedure used positive accelerations from the FLUSH analysis to calculate inertial loads which are then statically brought down to the basement. The maximum time history D-6 diagonal load in the FLUSH N-S run is 162k, while the static analysis has 257k. Thus $F = 1.6$.

The reinforced concrete (r/c) column on line T has a SSE margin of -0.19 under NS loading as discussed above. The axial load is about 500 psi, below the knee of the interactions curve, and so the column is expected to show the more ductile behavior of a beam. The modeling discussion relative to the D-6 diagonal load applies here, except that the soil in contact with the basement wall is expected to protect the substructure instead of amplifying the response.

Thus, in the existing configuration of building 109-NT, the overall factor of safety is 1.4 as shown on Table 4.27. Hence, the median failure level is given by $(1.4) \times 0.25g = 0.35g$. The uncertainties are as shown on Table 4.27.

If the proposed structural modifications to building 109-NT are performed and it is assumed that the modifications result in an increase in the SSE factor of safety from 0.51 to 1.2 (as is the case for building 182-N) then the overall factor of safety is increased to 3.3, and the resulting median acceleration level at failure is $(3.3) \times 0.25g = 0.824g$. The uncertainties will still be those given on Table 4.27. This is the fragility used for the "Upgrade Building 109NT Case" discussed later. These results however, are predicated on the assumption that the strength capacity is increased as assumed above.

4.5.3.3.9 Core Inlet Flanges and Piping

This section presents the seismic response and fragility characterization of the core inlet piping system under different earthquake levels. The

structural response was based on results of a series of analytical elastic analyses performed by Westinghouse-Hanford personnel (Ref. 45) in support of the N-Reactor seismic PRA.

An existing connector and piping analytical model was selected and used with best-estimate average (code-specified) pipe material properties. The model selected represents a group of N Reactor process piping systems. The line is designated 33-I-68 and was judged to envelope the behavior of all inlet connector piping systems outside the N Reactor graphite core.

The seismic load input consisted of thirty response spectra and structure differential motions. A total of thirty seismic cases were run, subdivided into three multiples of the Safe Shutdown Earthquake (1SSE, 2SSE, 3SSE).

Each analysis consisted of three independent orthogonal directions of excitation, i.e., X-north/south, Y-east/west, and Z-vertical. Also anchor terminals and supports of the piping system were attached to two different seismically isolated structures which required a multi-spectra seismic analysis. A schematic of line 33-I-68 is shown on Figure 4.39.

The seismic spectra used for this analysis differ somewhat from those used in the original N Reactor analyses to structurally qualify the systems. Load considerations have been added to include impact between adjacent structures along with differential motions between structure base and pipe location. For the north-south direction, effects of impact between the primary shield and the adjacent core have been included, as well as impact between the primary shield and the inlet/outlet shield for the east-west direction.

For all runs, a linear elastic dynamic multi-spectra seismic analysis was performed. To account for seismic differential motions, a static analysis considering respective seismic anchor movements was also performed. The piping analysis program 'PIPESD' which had been previously used for this model was employed for the above multi-spectra analysis. Each case had the following load sets and combinations:

Load Set/Combination	Title
1	Static Deadweight
2	Static Sum of Anchor Movements (Diff. Motions)
3	Thermal Load Set
4	Pressure
5	Dynamic Seismic (Multi-Spectra)
6	(Load 2 + Load 5)
7	(Load 1 + Load 4 + Load 5)

INLET CONNECTOR PIPE 33-I-6B

Directional (X, Y, and Z) components of seismic response quantities were determined for each seismic input. Dynamic responses were combined internally in the program in accordance with the "grouping method" specified in the NRC Regulatory Guide 1.92. To produce the total seismic response (in X-Y-Z components) for each case, spatial combination was carried out using the SRSS method. All other combination processes such as Load Set 6 and 7 were combined algebraically. (Note that the PIPESD Program performs the load combination process at member load level before code stress indices are applied.) Code Equation stresses (ASME and B31.1) were also included in the output for stress level evaluations. Data tabulations for the 1SSE, 2SSE, and 3SSE levels are given in Reference 45.

A comparison of maximum and minimum code stresses was prepared. Case runs from the 2SSE seismic level were used to determine the maximum and minimum seismic cases and also to locate the most critical node. Based on this comparison, Node 12 (a 9" bend radius elbow) was identified as the critical stress location. This is an elbow with 2.625 inches outside diameter and a thickness of 0.26 inches. The material is A106B carbon steel with a design/operating temperature of 600°F/400°F.

From the three sets of runs at the three levels, average seismic (dynamic) moments and average seismic anchor movement (SAM) moments were obtained. They were combined by SRSS to give the average net SRSS moment as shown below:

<u>Level</u>	<u>Average SRSS Moment</u>
1 SSE	27,900 in-lb
2 SSE	50,000 in-lb
3 SSE	64,200 in-lb

Random uncertainty for the ten time histories (at each level) resulted in a computed random uncertainty of 0.24 (beta-r) and the modeling uncertainty was judged to be 0.25 (beta-u).

The median failure moment was obtained from the piping fragility tables in Reference 30. These tables are referenced to a butt weld (6 inch, Sch. 160, A106B carbon steel at 70°F). Tables of scale factors relate other pipe sections (elbows, tees, etc.) at different temperatures and for different sizes and materials. Scaling was performed using scale factors for a butt weld. The result is a median failure moment of 121,000 in-pounds. This value was based on a ductility of 2.0 for Butt welds as specified in Reference 30. This value of ductility is appropriate for piping with no radiation embrittlement. The result was re-scaled to a median failure moment of 108,000 in-lb based on a reduced ductility of 1.6 which is felt to be more applicable to the N-Reactor core inlet piping penetrating the shield wall. The uncertainties specified in Reference 30 were used for the failure level (beta-r = 0.18 and beta-u = 0.33).

Finally, the median response and failure moments at each earthquake level were used (along with the corresponding sets of uncertainties) to calculate piping failure probabilities at each of the three levels as a function of peak ground acceleration. From these values, the median failure level in terms of PGA was determined to be 0.72g. The combined uncertainties are $\beta_r = 0.36$ and $\beta_u = 0.40$.

The analyses performed were limited to elastic pipe response to the various input spectra. Stresses for most load cases exceed the material yield strength (35000 psi for SA106 Gr. B) and some local yielding could be expected. Failure of the piping pressure boundary is unlikely, based upon recent studies, tests, and plant experience of typical system, as is reflected in the median acceleration to failure computed above which corresponds to the 3 SSE level.

4.6 Core Damage and Risk Computations

In the event of an earthquake or any other abnormal condition in a nuclear power plant, the plant safety systems act to bring the plant to a safe shutdown condition. In this step of the risk analysis process, we identify the possible paths that the nuclear plant would follow, given that an earthquake-related event has occurred which causes shutdown. These paths involve an initiating event and a success or failure designation for systems affecting the course of events, and are referred to as accident sequences.

4.6.1 Initiating Events

The seismic analysis performed for N Reactor is based on the same set of LOCA and transient event trees developed for the internal event analyses of the plant, plus two additional seismically-induced structural failure initiators leading directly to core damage. The initiating events considered are:

- a. Process Tube Rupture (ECCS ineffective)
- b. Large LOCA
- c. Small LOCA
- d. Buildings 182/109NT
- e. Transient Type 1 (LOSP and Loss of Electrical Train A)
- f. General Transient (PCS initially available)

Note that one structure failure has been added to the list of initiating events. The building 182-N/109-NT failure was added as an initiating event since it leads directly to an accident sequence leading to core damage, and thus was included explicitly in the hierarchy of initiating events.

The pressure tube rupture event was computed based on the probability of failure of the shield wall, the inlet core piping or the walls of the 105-N building. Building 109-N failure is assumed to cause a large LOCA. The frequency for the small LOCA initiating event was computed based on the failure of piping in the reactor coolant loop. The frequency for these pipe failures was generated from the calculations of piping failures in the SSMRP Zion analysis (Ref. 22). This distribution is shown in Figure 4.40. The independent variable for this figure is peak ground acceleration, with a random variability of 0.25. This distribution was input in log normal form for the analysis.

The Type 1 transient initiating event was based on the probabilities of loss of offsite power (LOSP), failure of the 4KV and 13.8 busses, and structural failure of Buildings 151, 153 and 184. All of these result in loss of A train emergency AC power.

Figure 4.40 Frequency of Pipe Breaks Causing Small
LOCAs Derived From SSMRP Piping Calculations
(Reference 22)

In computing the frequency of the initiating events, a hierarchy between them must be established. The order of this hierarchy is such that, if one initiating event occurs, the occurrence of other initiating events further down the hierarchy is of no consequence. The hierarchy of initiating events is shown in Figure 4.41. Thus, for example, if a large LOCA occurs, it is not of concern if a small LOCA or transient also occurs. The most serious initiating event is the process tube rupture event since it is beyond the ECCS design basis. The probability of the LLOCA event is then computed as the probability of the anchorage failure times the complement of the PTR initiating event, and similarly, for the SLOCA and T1 events. The building 182-N/109-NT failures were put above the T1 (LOSP) transient inasmuch as these building failures result in loss of the ultimate heat sink for the plant. Implicit in the hierarchy definition is the requirement that events in the hierarchy identified as being more serious than a given initiating event cannot occur in the accident sequence for that event. For example, LOSP can occur as a basic event in any of the LOCA sequences or building 182-N/109-NT failure sequences, but cannot occur as a basic event in the general transient accident sequences.

With the hierarchy established, the general transient initiating event probability is computed from the condition that the sum of the initiating event probabilities considered must be unity. The hypothesis is that, given an earthquake of reasonable size, at least one of the initiating events will occur. If no other failures or signals SCRAM the plant, N Reactor has seismoscopes set to SCRAM the plant at ground accelerations of 0.007g. If the seismoscopes fail, a manual trip would be initiated.

Numerical values for the initiating events at various earthquake levels are given in Section 4.6.5. Numerical values for the parameters of the fitted distributions are listed in the fragility file (FRAG) in Appendix E.

4.6.2 Event Trees

The event trees developed for the internal event analyses were used directly, so as to be able to compare the final core damage frequencies due to seismic and internal events on a common basis. The internal event trees used for all external events analyses were presented in Section 2.2. They are repeated here for ease of subsequent discussion as Figures 4.42 through 4.44.

Finally, assignment of the accident sequences and their cut sets to the different damage states was performed by examination of the cut sets in both the accident sequences and the containment system sequences, as will be described later.

Figure 4.41 Initiating Event Hierarchy Tree

LARGE LOCA INITIATOR	REACTIVITY CONTROL	ECCS STARTS AND FUNCTIONS	ECCS FLOW OCCURS IN ALL 16 RISERS	ECCS CONTINUES TO FUNCTION	GSCS COOLS FUEL VIA THE GRAPHITE	Endstates	Sequence Designator	Fuel Damage Sequence Number
L	D	A	A2	A1	C	OK	L	
						OK	LA1	
						Z1 (DEL)	LA1C	L1
						W	LA2	L2
						X	LA2C	L3
						W	LA2A1	L4
						Z1 (DEL)	LA2A1C	L5
						Y	LA	L6
						Z	LAC	L7
						Z	LD	L8

Figure 4.42 Large LOCA Event Tree

Figure 4.43 Small LOCA Event Tree

Figure 4.44 General Transient (PCS initially available) Event Tree

4.6.3 Failure Modes of Safety Systems

To determine failure modes for the plant safety systems, fault tree methodology is used. This methodology systematically identifies all groups of components in a system which, if failed simultaneously, would result in failure of that system.

Construction of a fault tree begins by identifying the immediate causes of system failure. Each of these causes is then examined for more fundamental causes, until one has constructed a downward branching tree, at the bottom of which are failures not further reducible, i.e., failures of mechanical or electrical components due to all causes such as structural failure, human error, maintenance outage, etc. These lowest order failures on the fault tree are called basic events. Failures of basic events due to seismic ground motions, random failures, human error, and test and maintenance outages are included in the seismic analyses.

The main difference between an internal event fault tree for a safety system and an external event fault tree is that consideration must be given to the physical location of the components, because the physical location determines to what extent secondary failures become important. Examples of this would be secondary failures due to local masonry wall collapse or due to a high temperature/steam environment from a broken steam line. Hence, in performing the seismic analyses, the locations of all important pieces of equipment must be determined from the general arrangement drawings for the plant, and then a systematic examination for secondary failure possibilities is made during the plant walkdown.

As stated earlier, the internal event PRA fault trees form the basis for the fault trees used in the seismic analysis. This allows for a consistent level of detail between internal and external event analyses, and assures the consistent inclusion of random and test/maintenance outage unavailabilities in the seismic analysis.

Since the internal event fault trees already exist and seismic failure modes are to be added, one must modify the internal event fault trees to include:

- a. Local structural failures (block walls, cranes, etc.), and
- b. Failure of critical passive components (tanks, cable tray failures, and pipes) often missing in internal events analysis.

This is accomplished in several ways. First, the secondary or passive failure event can be added directly to the fault tree structure and the "gate" definition data file modified. Alternatively, the fragility definition of a relatively strong component on the tree may be redefined in terms of the (relatively weaker) associated secondary failure. Finally, events globally affecting a safety system or an accident sequence (such as building failure or liquefaction) can be added directly to the boolean expression.

4.6.4 Accident Sequence Evaluation

Accident sequence probabilities are used in determining the frequencies of core damage and of radioactive release for a given release category. Core damage frequency is defined as the sum of the frequencies of all accident sequences leading to core damage.

4.6.4.1 General Considerations

Each accident sequence consists of groups of events (successes or failures of safety systems) which must occur together. The failure of each safety system can be represented in terms of minimal cut sets, which are groups of component failure which will cause the safety system to fail. These cut sets and the accident sequences are combined together so that every accident sequence can be expressed in a Boolean expression of the form

$$ACC_j = IE_j [C_1C_2C_3 \text{ or } C_4C_5 \text{ or } \dots \text{ or } C_1C_jC_k]$$

in which IE_j is the initiating event and the C_i are basic events (i.e., failure of individual components) identified on the system fault trees. If at least one of the component failure groups $C_1C_jC_k$ occurs, then the accident sequence occurs. Computation of each accident sequence probability consists of determining the probability of each cut set, and then combining them to get the accident sequence probability.

Each basic event seismic failure probability is computed assuming that the response and fragility distributions are in log normal form. Calculations in the SSMRP showed that responses were reasonably fit by log normal distributions. The limited data on fragilities can be fit with log normal distributions as well as any other type. Hence, for convenience the log normal distribution is used for both. The equation used to calculate seismic failure frequencies is given as:

$$P_f = \Phi \left[\frac{\ln (m_R / m_F)}{\sqrt{\beta_{FR}^2 + \beta_{RR}^2}} \right]$$

where Φ is the standard normal cumulative distribution function, m_R and m_F are the response and fragility medians, and β_{RR} and β_{FR} are the random uncertainties in the response and fragility distributions, respectively. Note that the use of log normal distributions is not essential to the calculation process used in these calculations, and, in fact, any arbitrary pair of distributions could be used for the responses and fragilities provided they are physically meaningful.

When the individual basic failure events in a cut set $C_1C_jC_k$ are not independent, correlation between the basic events must be explicitly included. When only two of the basic events are correlated the joint probabilities may be computed directly by the use of tables. When more than two basic events in a cut set are correlated, numerical multiple integration may be used (such as performed by the SEISIM code developed in the SSMRP, Ref. 22).

Finally, the accident sequences defined above are a function of peak ground acceleration, and as such, are conditional on the hazard curve. They are subsequently unconditioned by integrating these sequences over the hazard curve as described subsequently.

4.6.4.2 Accident Sequence Quantification

Quantification of the accident sequences is a multi-step procedure involving several levels of screening. In the first step, the SETS code (Ref. 48) is used to evaluate all potential accident sequences using point estimate input screening values for all the seismic failure events (and using the internal events point estimate failure values for all random events). The same fault trees used by the internal events analysis are solved here with additions as noted in Section 4.6.3. The seismic screening values are taken as some conservative estimate, usually the component failure probabilities evaluated at three times the SSE. A dual probabilistic culling criterion is used in this culling process. This dual criterion is used in recognition of the fact that potentially large correlations can exist between basic events in the same cut set due to the pervasive nature of the seismic input motion. The result of this screening step is a reduced set of Boolean equations describing each of the safety and support systems.

In the second step, again utilizing the SETS code, (Ref. 46) these Boolean equations are merged together to form the accident sequences, again as defined for the internal events analysis. At this stage, truncation is performed based both on the order of the cut sets as well as the probability of the cut sets. The result of this step is Boolean equations describing each accident sequence and containing all the important seismic and random failure events.

The final step involves the actual quantification of the accident sequences. These accident sequence expressions are utilized both to compute point estimates of the accident sequence frequencies and to perform the uncertainty analysis calculations. A cross reference table (XREF) is set up which relates each component to a component ID number, its random point estimate and error factor value, and to its associated seismic fragility category and seismic response category. This cross reference table thus provides all the information required to compute the probability of failure of any basic event (random or seismic or combined) at any peak ground acceleration level. The cross reference table for N Reactor is presented in Appendix E.

Prior to performing the complete uncertainty analysis, a point estimate calculation is performed to gain insight into the dominant sequences, components and earthquake levels contributing to the core damage frequency. The point estimate computed is the "mean" point estimate. This is computed by using the mean random failure frequencies, the mean seismic hazard curve, and using mean values for the seismic failure event frequencies. This mean point estimate will be seen to be nearly equal to the exact mean value of accident sequence and core damage frequency as obtained in the uncertainty analysis. Again, this mean value point estimate is useful in understanding the contributions of the various basic events to the total frequencies and to understanding the contributions to the total uncertainty bands. Because mean values do add to yield the mean value of each accident sequence (conditional on the hazard) the only difference between the true mean and the mean point estimate has to do with integrating the conditional accident frequencies over the hazard curve. Experience has shown, however, that the difference between these two is very small.

Finally, a complete uncertainty analysis is performed on the dominant accident sequences (and on the dominant cut sets in each accident sequence) as determined in the point estimate screening evaluations. A Monte Carlo analysis was used for the N-Reactor studies. Thus, the expression for the unconditional accident sequence frequencies (and for core damage frequency) shown as below:

$$ACC_j = \int P(ACC_j, PGA) f_{eq}(PGA) d(PGA)$$

where

$P(ACC_j, PGA)$ is the conditional accident sequence frequency as a function of PGA, and

$f_{eq}(PGA)$ is the probability density function for the hazard curve,

is randomly sampled varying the hazard curve parameters, the random failure frequencies, and the seismic response and fragility parameters. From the accumulated values of accident sequence frequencies and core damage frequency, exact statistics on the distributions are directly obtained.

Note that in performing the uncertainty analyses, full correlation between random samples taken from the same response category and from the same fragility category was enforced. In effect, this implies that our lack of knowledge about the failure characteristics for all components in a given generic category is the same, and if we are wrong for one component, then we are wrong for all components by the same amount.

4.6.5 N-Reactor Core Damage Frequency Results

This section presents the results of the seismic risk analysis for the N-Reactor nuclear power plant. The base case is our best estimate of the current configuration of the plant and its emergency procedures. In particular, the seismic component failure probabilities were taken from the generic fragility data base (Table 4.19) the site-specific component fragilities from Table 4.20, and the building fragilities as given in Table 4.22. The "Upgraded Building 109-NT Case" used the same input with the exception that the building 109-NT fragility median was increased from the base case value (0.35g) to the upgraded value of 0.824g as determined in Section 4.5.3.3.8. As described earlier, a six initiating events and three event trees were used for the seismic risk analysis.

A total of 13 accident sequences survived the (conservative) screening process. These 13 sequences are presented in Table 4.29 along with identification of the boolean sequences that were solved for each accident sequence. (The number of booleans solved using the SETS code is less than the number of accident sequences because several accident sequences may utilize the same boolean expression even though the initiating event may be different.) Also identified on this table are the complement expressions which must be included in the numerical sequence quantification. (These are important at high PGA levels at which success probabilities may be significantly less than unity.) The analytical equations used for calculating the boolean sequences and the complement factors are presented in Appendix E. Table 4.30 describes the abbreviations used for the accident sequences in Table 4.29. Table 4.31 describes how the site-specific building and component failures were incorporated in the systems logic models.

These 13 non-negligible accident sequences resulting from the screening process were fully re-quantified using best-estimate random failures mean probabilities and best estimate seismic fragilities and responses plus associated variabilities. Based on this final quantification, five dominant sequences were identified. These dominant sequences are:

<u>Accident Sequence</u>	<u>Base Case</u>	<u>Upgrade 109NT Case</u>
BLDGS 182 AND 109-NT	63%	11%
PTR	17%	34%
LOSP T8	12%	43%
SLOCA 9	4%	2%
SLOCA 8	3%	7%

The percentage contributions were taken from the Monte Carlo uncertainty results. The total mean core damage frequency for the N Reactor base case was computed to be $4.6E-5$ per year. The relative contributions of the accident sequences are shown in Table 4.32. Relative importance of the basic events to these results are presented in the point estimate

Table 4.29
Seismic Accident Sequences

<u>Accident Sequence</u>	<u>Boolean Expression</u>	<u>Boolean Equation</u>	<u>Complement Factor</u>
Process Tube Rupture	N/A		
1. PTR			
<u>Large LOCA</u>			
2. LLOCA-6	$A \bar{C}$	BOOL 3	$\bar{C}$
3. LLOCA-7	$A C$	BOOL 1	
<u>Small LOCA</u>			
4. SLOCA-4	$S_2 G \bar{A} A_2 \bar{A}_1 \bar{C}$	BOOL 7	$\bar{A} \bar{A}_1 \bar{C}$
5. SLOCA-8	$S_2 G A \bar{C}$	BOOL 4	$\bar{C}$
6. SLOCA-9	$S_2 G A C$	BOOL 2 and 11	
<u>Building 182/109NT</u>			
7. Bldg 182/109NT			
<u>LOSP Transients</u>			
8. LOSP-T4	$LOSP G \bar{A} A_2 \bar{A}_1 \bar{C}$	BOOL 7	$\bar{A} \bar{A}_1 \bar{C}$
9. LOSP-T8	$LOSP G A \bar{C}$	BOOL 4	$\bar{C}$
10. LOSP-T9	$LOSP G A C$	BOOL 2	
<u>General Transients</u>			
11. TRANS-4	$TRANS G \bar{A} A_2 \bar{A}_1 \bar{C}$	BOOL 7	$\bar{A} \bar{A}_1 \bar{C}$
12. TRANS-8	$TRANS G A \bar{C}$	BOOL 4	$\bar{C}$
13. TRANS-9	$TRANS G A C$	BOOL 2	

Table 4.30

Safety Systems Nomenclature

Systems

- G Primary/Secondary Cooling Systems (0-2 hrs)
- G₁ Primary/Secondary Cooling Systems (2-24 hrs)
- A Emergency Core Cooling System (0-2 hrs)
- A₁ Emergency Core Cooling System (2-24 hrs)
- A₂ Failure of ECCS check valves leading to damage to 1/16 core
- C Graphite Shield Cooling System

Initiating Events

- PTR Process Tube Rupture Initiator (ECCS ineffective)
- A Large LOCA Initiator
- S₂ Small LOCA Initiator
- TRANS General Transient Initiator

Table 4.31

Site-Specific Building/Tank/Component Failures and Their
Impact on Systems

<u>Failure</u>	<u>Effect</u>	<u>Impact</u>
Building 105-N	Fails Reactor/ECCS	Causes PTR event
Building 109-N	Fails SGS, PZR, Etc.	Causes LLOCA event
Primary Shield Wall or Inlet Piping	Fall Pressure Boundary and ECCS	Causes PTR event
Building 182-N	Fails HLDPs (A) Fails Support Pumps for PCS (G). Fails makeup to C.	Transient Initiator leading to GAC directly
Building 109-NT	Fails G,A,C, due to system's cable failure	Same as Bldg 182 N
Building 181-N	Fails LLDPs (A)	Not Initiator. Fails A in L6, L7, S8, S9, T8, T9.
Building 151	Fails A Bus and hence G.	Source of LOSP initiator. Acts like LOSP in all sequences above LOSP.
Buildings 153 and 184	Fail both A & B Busses, and thus all AC power. Also fails PCS/SCS due to failures in instru- ment air system.	Acts as common cause 4KV bus failure.
Silo	Fails ECCS	Single failure of A in any sequence.
De-Min Tank	If diversion or fail to switch to silo, fails ECCS	Affects only cut sets.
13.8KV A Bus	Sliding Failure	Causes LOSP
13.8KV B Bus	Sliding Failure	Fails B Elec Train
4KV A Bus	Sliding Failure	Causes LOSP
4KV B Bus	Sliding Failure	Fails B Elec Train

Table 4.32

Accident Sequence and
Total Core Damage Mean Frequencies

<u>Accident Sequence</u>	<u>Mean Frequency (per year)</u>	
	<u>Base Case</u>	<u>Upgrade 109-NT Case</u>
1. PTR	7.43E-6	7.43E-6
2. LLOCA-6	3.16E-9	3.16E-9
3. LLOCA-7	5.76E-9	2.82E-9
4. SLOCA-4	1.42E-8	1.42E-8
5. SLOCA-8	1.51E-6	1.51E-6
6. SLOCA-9	2.04E-6	3.16E-7
7. Bldg. 182/109NT	2.86E-5	1.57E-6
8. LOSP-T4	3.04E-7	3.74E-7
9. LOSP-T8	6.25E-6	1.09E-5
10. LOSP-T9	4.23E-8	1.10E-7
11. TRANS-4	4.28E-9	4.51E-9
12. TRANS-8	7.00E-9	9.17E-9
13. TRANS-9	2.10E-11	2.78E-11
Total	4.62E-5 per yr	2.22E-5 per yr

results presented below. Percentiles of the distributions obtained for the total core damage frequency and for each of the accident sequences are given in Table 4.33. Note that the range factor (defined as the ratio of the 95th percentile to the 50th percentile of the distribution) on the total core damage frequency was found to be 39.

For the upgraded building 109-NT case, the total core damage frequency was found to be reduced to $2.2\text{E-}5$ per year. The relative contributions of the individual accident sequences to the total are also shown on Table 4.32. The range factor for this case was computed to be 49.

4.6.5.1 Description of Accident Sequences

To understand the dominant accident sequences it must be recalled that there are essentially three mitigating systems identified on the various event trees. System G is the primary and secondary coolant systems. The designator G denotes successful functioning in the first two hours whereas G_1 denotes successful functioning of the system in the 2-24 hour time frame. Similarly, system A is the emergency core cooling system (ECCS) operating successfully in the zero to 2 hour timeframe. Designator A_1 is the same ECCS system in a 2-24 hour timeframe. The designator A_2 denotes failure of a single check valve on the inlet side of the core resulting in failure of 1/16 of the core. Finally, system C denotes the graphite shield cooling system (GSCS). The GSCS system is assumed to be effective in removing decay heat only following 2 hours after shutdown. Whether or not the GSCS system fails between zero and 2 hours after shutdown only determines the release category for the off-site risk calculations. That is, in each case, core damage is assumed to occur whether or not the GSCS system works or fails. Finally, it is recalled from Chapter 2 that for large LOCA situations no credit is taken for the primary and secondary cooling systems.

The base case dominant accident sequence results from the failure of buildings 182-N or 109-NT. As noted on Table 4.31 this leads directly to accident sequence GAC. No other failures are required to cause this transient accident sequence. Building 182-N failure is assumed to fail the PCS/SCS due to numerous support pumps for this system which are located in this building. Similarly, the ECCS system fails because the high-lift diesel pumps are located in building 182-N. The GSCS fails because its makeup pumps are located in Building 182-N. In the case of building 109-NT, failure of the control cabling for all frontline systems is assumed to occur given building 109-NT failure, which causes the GAC sequence to occur directly. For the base case, the median fragility for Bldg 109NT is significantly lower than that for building 182-N, and hence is the dominant contributor to this sequence.

The second largest base case core damage frequency contributor is the process tube rupture sequence. This arises due to failure of the shield wall or the inlet process tube piping as discussed in 4.5. The shield wall is essentially pin supported at the inlet and outlet end and gross failure of that wall would result directly in shearing a number of the

Table 4.33

Base Case Accident Sequence Frequency
Distribution Percentiles

<u>Sequence</u>	<u>Mean</u>	<u>Var</u>	<u>5%</u>	<u>50%</u>	<u>95%</u>
Process Tube Rupture	7.43E-06	1.35E-09	3.19E-05	2.81E-07	5.24E-12
LLOCA-6	3.16E-09	1.36E-15	2.30E-09	7.67E-13	1.16E-13
LLOCA-7	5.76E-09	4.73E-15	3.26E-09	1.09E-12	9.98E-14
SLOCA-4	1.42E-08	6.51E-15	4.38E-08	8.36E-10	5.68E-12
SLOCA-8	1.51E-06	1.48E-10	2.98E-06	4.92E-08	3.73E-11
SLOCA-9	2.04E-06	2.00E-10	5.95E-06	1.01E-07	1.71E-10
Bldg's 182-N/109-NT	2.86E-05	2.93E-08	1.09E-04	2.28E-06	2.00E-08
LOSP-T4	3.04E-07	1.86E-12	1.02E-06	4.50E-08	1.56E-09
LOSP-T8	6.25E-06	2.35E-09	1.70E-05	4.18E-07	1.33E-08
LOSP-T9	4.23E-08	1.36E-13	6.74E-08	8.29E-10	1.54E-12
TRANS-4	4.28E-09	1.52E-15	9.41E-09	1.48E-10	8.52E-13
TRANS-8	7.00E-09	1.14E-14	6.78E-09	3.88E-11	3.13E-13
TRANS-9	2.10E-11	1.39E-19	1.22E-11	1.34E-13	4.21E-15
Total Core Damage	4.62E-05	6.25E-08	1.66E-04	4.24E-06	9.18E-08

inlet or outlet pipes thus resulting in an uncoolable situation. The inlet core piping is stressed due both to inertia-induced moments as well as due to seismic anchor relative motions. No other failures are required to cause this accident sequence.

Two of the dominant sequences (LOSP T8 and SLOCA 8) result from the same logical combination of cut sets and component failures (contained in boolean expression BOOL 4 in Appendix E). This follows from the fact that the general transients, the loss of offsite power transients and the small LOCA transients all use the same general event tree although different success criteria are sometimes used. Thus, the BOOL-4 expression applies to a number of different sequences. In each case, the sequence involves GA-C system failures. The dominant cut sets contributing to these accident sequences consists of failure of the silo (which fails the ECCS system) in conjunction with electrical bus failures (both 13.8KV and 4KV) which fail the primary and secondary cooling systems. In these sequences, the GSCS system is not failed. The failure of the PCS/SCS is caused by a dependency on instrument air which in turn requires power from 4KV busses. These component failure combinations are far and away the dominant contributors to the LOSP T8 and SLOCA 8 accident sequences.

The remaining dominant sequence is SLOCA 9. This involves the system failures G, A, and C. Booleans BOOL 2 and BOOL 11 contain the cut sets resulting in this GAC sequence. These involve LOSP or valves and failure of the silo. In addition, failure of buildings 182-N and 109-NT cause this sequence (in conjunction with the SLOCA initiator) and these failures are the dominant contributors to the sequence.

In the upgraded building 109-NT case, the sequences involving primarily building 109-NT failure are significantly reduced (SLOCA T9 and Sequence 7 which has the building 109-NT failure as part of the initiator) and since the initiating event frequencies add to unity, the LOSP and general transient initiating event frequencies increase as the building 182-N/109-NT initiator decreases.

4.6.5.2 Base Case Mean Point Estimate

As described earlier, this point estimate is based on using the mean values for all variables. The mean initiating event frequencies at different PGA values are given in Table 4.34. As can be seen, at the lower earthquake levels the transient sequence initiating events dominate, and as the earthquake acceleration level increases, the LOSP and building 182-N/109-NT initiators increase until, finally, at the higher earthquake levels, there is a significant contribution from the process tube rupture (PTR) event. Also note that, at each earthquake level, the initiating events sum to 1.0. Values of the dominant accident sequence conditional frequencies at various earthquake levels are presented in Table 4.35. These are the values that are integrated over the hazard curve to obtain the unconditional accident sequence frequencies.

Table 4.36 presents the mean core damage contributions at seven intervals

Table 4.34

Initiating Event Frequencies - Mean Point Estimate - Base Case

	0.1g	0.2g	0.3g	0.4	0.5g	0.6g	0.7g
PTR	6.7E-05	7.2E-03	5.6E-02	1.8E-01	3.5E-01	5.4E-01	7.1E-01
LLOCA	1.0E-10	2.4E-07	2.0E-05	2.6E-04	1.2E-03	3.2E-03	5.3E-03
SLOCA	4.6E-04	1.1E-02	4.2E-02	8.2E-02	1.1E-01	1.1E-01	9.3E-02
182/109NT	5.0E-03	1.2E-01	3.4E-01	4.6E-01	4.2E-01	3.0E-01	1.8E-01
LOSP	1.4E-01	7.3E-01	5.6E-01	2.8E-01	1.2E-01	4.0E-02	1.2E-02
TRANS	8.5E-01	1.3E-01	3.3E-03	3.8E-05	3.2E-07	1.9E-08	0.0E-01

Table 4.35

Conditional Accident Sequence Frequencies - Mean Point Estimate - Base Case

	0.1g	0.2g	0.3g	0.4g	0.5g	0.6g	0.7g
PTR	6.7E-05	7.2E-03	5.6E-02	1.8E-01	3.5E-01	5.4E-01	7.1E-01
LLOCA-6	1.0E-09	7.7E-09	2.2E-06	7.4E-05	6.1E-04	2.1E-03	4.2E-03
LLOCA-7	1.0E-09	3.0E-08	7.4E-06	1.6E-04	9.7E-04	2.8E-03	5.0E-03
SLOCA-4	1.9E-07	2.7E-05	1.4E-04	3.2E-04	4.7E-04	5.2E-04	4.5E-04
SLOCA-8	1.2E-06	2.9E-04	8.1E-03	4.8E-02	9.8E-02	1.1E-01	9.3E-02
SLOCA-9	2.3E-06	1.3E-03	1.6E-02	5.1E-02	8.6E-02	9.9E-02	8.8E-02
182/109NT	5.0E-03	1.2E-01	3.4E-01	4.6E-01	4.2E-01	3.0E-01	1.8E-01
LOSP-T4	3.9E-04	2.0E-03	1.5E-03	7.8E-04	3.2E-04	1.1E-04	3.3E-05
LOSP-T8	2.8E-03	2.4E-02	6.3E-02	8.1E-02	5.7E-02	2.7E-02	9.6E-03
LOSP-T9	1.3E-07	1.1E-04	8.5E-04	1.4E-03	1.1E-03	5.4E-04	2.0E-04
TRANS-4	1.1E-05	2.7E-05	2.1E-06	4.3E-08	1.0E-09	1.0E-09	1.0E-09
TRANS-8	1.0E-06	2.7E-04	1.6E-04	9.2E-06	1.8E-07	1.6E-08	1.0E-09
TRANS-9	1.0E-09	6.0E-07	4.4E-07	2.7E-08	1.0E-09	1.0E-09	1.0E-09

Table 4.36

Total Accident Sequence Frequency Increments - Mean Point Estimate - Base Case

	0.05g/ 0.15g	0.15g/ 0.25g	0.25g/ 0.35g	0.35g/ 0.45g	0.45g/ 0.55g	0.55g/ 0.65g	0.65g/ 0.75g	Total
PTR	1.3E-08	5.5E-07	1.7E-06	2.1E-06	1.7E-06	1.0E-06	5.3E-07	7.6E-06
LLOCA-6	1.9E-13	5.8E-13	6.7E-11	8.9E-10	2.9E-09	4.0E-09	3.2E-09	1.1E-08
LLOCA-7	1.9E-13	2.3E-12	2.2E-10	1.9E-09	4.7E-09	5.3E-09	3.7E-09	1.6E-08
SLOCA-4	3.6E-11	2.1E-09	4.3E-09	3.8E-09	2.3E-09	9.8E-10	3.4E-10	1.4E-08
SLOCA-8	2.2E-10	2.2E-08	2.5E-07	5.8E-07	4.7E-07	2.1E-07	7.0E-08	1.6E-06
SLOCA-9	4.4E-10	1.0E-07	4.8E-07	6.1E-07	4.2E-07	1.9E-07	6.6E-08	1.9E-06
182/109NT	9.5E-07	9.4E-06	1.0E-05	5.5E-06	2.0E-06	5.7E-07	1.4E-07	2.9E-05
LOSP-T4	7.5E-08	1.5E-07	4.6E-08	9.4E-09	1.5E-09	2.1E-10	2.4E-11	2.9E-07
LOSP-T8	5.4E-07	1.8E-06	1.9E-06	9.8E-07	2.8E-07	5.1E-08	7.2E-09	5.6E-06
LOSP-T9	2.6E-11	8.6E-09	2.6E-08	1.7E-08	5.2E-09	1.0E-09	1.5E-10	5.7E-08
TRANS-4	2.0E-09	2.1E-09	6.5E-11	5.2E-13	4.8E-15	1.9E-15	7.5E-16	4.2E-09
TRANS-8	1.9E-10	2.1E-08	5.0E-09	1.1E-10	8.7E-13	3.0E-14	7.5E-16	2.6E-08
TRANS-9	1.9E-13	4.6E-11	1.3E-11	3.2E-13	4.8E-15	1.9E-15	7.5E-16	6.0E-11
	1.6E-06	1.2E-05	1.5E-05	9.8E-06	4.9E-06	2.1E-06	8.2E-07	4.60E-05

over the hazard curve for each accident sequence. Integration over the mean hazard curve was performed from 0.05 g to 0.75 g and integration increments of 0.1 g were utilized. The right hand column presents the total contribution of each accident sequence to the total core damage frequency of $4.60\text{E-}5$.

An important thing to note from Table 4.36 is the sum of the accident sequence contributions at each earthquake level, as shown at the bottom of each column on the table. The contributions are seen to be small at the first increment, increasing to a maximum and then decreasing at higher earthquake levels. This indicates that the bulk of the risk is occurring in the range of 0.25 g to 0.55 g which roughly corresponds to the range of 1-2 SSE. Further, this shows that the bulk of the risk has been captured by integrating over the range 0.05 g to 0.75 g:

4.6.5.3 Upgrade Building 109-NT Case Mean Point Estimate

Tables 4.37 through 4.39 present similar results for the upgraded building 109-NT case. In this case a total core damage frequency of $2.18\text{E-}5$ was computed. This is very close to the Monte Carlo estimate of mean core damage frequency of $2.22\text{E-}5$ computed using the same equations in the uncertainty analysis. Similar comments with respect to the variation of initiating event frequencies and accident sequence frequencies with earthquake level as described for the base case apply.

4.6.6 Importance Studies

4.6.6.1 Basic Event Importance to Mean Values

The importance of the each basic seismic failure event was evaluated by (sequentially) setting its seismic failure probability to zero in the mean point estimate calculation, which gives a measure of the net reduction in core damage frequency that would occur if that component could never fail due to seismic shaking. Results of these calculations are shown in Table 4.40. It can be seen that, for the base case, the largest reduction occurs for building 109-NT due to its importance in the accident sequences and its relatively low median fragility. The second most important component is the silo. The failure of the silo results in failure of the water sources for the high lift diesel driven pumps which fails ECCS directly. The third most important component is the ceramic insulators whose failure leads to the LOSP sequences. Finally, the fourth most important is the core inlet piping whose failure leads directly to a process tube rupture event beyond the capability of the ECCS. All other components and structures had risk reduction potentials of less than 1 percent.

The picture is the same for the upgraded building 109-NT case, except that now the building 109-NT failure is much less important. The silo failure is the most important component failure due to the large number of accident sequences in which it occurs as the cause of ECCS failure. The core inlet piping and the ceramic insulators are now also relatively more important.

Table 4.37

Initiating Event Frequencies - Mean Point Estimate - Bldg 109-NT Upgrade Case

	0.1g	0.2g	0.3g	0.4g	0.5g	0.6g	0.7g
PTR	6.7E-05	7.2E-03	5.6E-02	1.8E-01	3.5E-01	5.4E-01	7.1E-01
LLOCA	1.0E-10	2.4E-07	2.0E-05	2.6E-04	1.2E-03	3.2E-03	5.3E-03
SLOCA	4.6E-04	1.1E-02	4.2E-02	8.2E-02	1.1E-01	1.1E-01	9.3E-02
182/109NT	7.5E-06	2.0E-03	2.1E-02	6.6E-02	1.1E-01	1.2E-01	9.8E-02
LOSP	1.4E-01	8.4E-01	8.8E-01	6.8E-01	4.3E-01	2.2E-01	9.8E-02
TRANS	8.6E-01	1.4E-01	5.1E-03	9.0E-05	1.1E-06	1.5E-08	0.0E-01

Table 4.38

Conditional Accident Sequence Frequencies - Mean Point Estimate -
Bldg 109-NT Upgrade Case

	0.1g	0.2g	0.3g	0.4g	0.5g	0.6g	0.7g
PTR	6.7E-05	7.2E-03	5.6E-02	1.8E-01	3.5E-01	5.4E-01	7.1E-01
LLOCA-6	1.0E-09	7.7E-09	2.2E-06	7.4E-05	6.1E-04	2.1E-03	4.2E-03
LLOCA-7	1.0E-09	1.0E-09	4.8E-07	2.4E-05	2.6E-04	1.1E-03	2.7E-03
SLOCA-4	1.9E-07	2.7E-05	1.4E-04	3.2E-04	4.7E-04	5.2E-04	4.5E-04
SLOCA-8	1.2E-06	2.9E-04	8.1E-03	4.8E-02	9.8E-02	1.1E-01	9.3E-02
SLOCA-9	3.5E-09	2.3E-05	1.0E-03	7.7E-03	2.3E-02	4.0E-02	4.7E-02
182/109NT	7.5E-06	2.0E-03	2.1E-02	6.6E-02	1.1E-01	1.2E-01	9.8E-02
LOSP-T4	3.9E-04	2.3E-03	2.4E-03	1.8E-03	1.2E-03	6.1E-04	2.7E-04
LOSP-T8	2.8E-03	2.7E-02	9.9E-02	1.9E-01	2.1E-01	1.5E-01	7.9E-02
LOSP-T9	1.3E-07	1.3E-04	1.3E-03	3.3E-03	3.9E-03	3.0E-03	1.7E-03
TRANS-4	1.1E-05	3.1E-05	3.4E-06	1.0E-07	1.8E-09	1.0E-09	1.0E-09
TRANS-8	1.0E-06	3.1E-04	2.6E-04	2.2E-05	6.4E-07	1.2E-08	1.0E-09
TRANS-9	1.0E-09	6.9E-07	6.9E-07	6.3E-08	2.0E-09	1.0E-09	1.0E-09

Table 4.39

Total Accident Sequence Frequency Increments - Mean Point Estimate -

Bldg 109-NT Upgrade Case

	0.05g/ 0.15g		0.15g/ 0.25g		0.25g/ 0.35g		0.35g/ 0.45g		0.45g/ 0.55g		0.55g/ 0.65g		0.65g/ 0.75g		Total
PTR	1.3E-08	5.5E-07	1.7E-06	2.1E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	1.7E-06	7.6E-06
LLOCA-6	1.9E-13	5.8E-13	6.7E-11	8.9E-10	2.9E-09	2.9E-09	2.9E-09	2.9E-09	2.9E-09	2.9E-09	2.9E-09	2.9E-09	2.9E-09	2.9E-09	1.1E-08
LLOCA-7	1.9E-13	7.6E-14	1.5E-11	2.9E-10	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.3E-09	1.3E-09	5.8E-09
SLOCA-4	3.6E-11	2.1E-09	4.3E-09	3.8E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	2.3E-09	1.4E-08
SLOCA-8	2.2E-10	2.2E-08	2.5E-07	5.8E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	4.7E-07	1.6E-06
SLOCA-9	6.6E-13	1.8E-09	3.1E-08	9.3E-08	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07	1.1E-07	3.5E-07
182/109NT	1.4E-09	1.5E-07	6.4E-07	8.0E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	5.3E-07	2.4E-06
LOSP-T4	7.5E-08	1.7E-07	7.3E-08	2.2E-08	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09	5.6E-09	3.5E-07
LOSP-T8	5.5E-07	2.1E-06	3.0E-06	2.3E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	1.0E-06	9.3E-06
LOSP-T9	2.6E-11	9.8E-09	4.0E-08	3.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.9E-08	1.2E-07
TRANS-4	2.0E-09	2.4E-09	1.0E-10	1.2E-12	8.5E-15	8.5E-15	8.5E-15	8.5E-15	8.5E-15	8.5E-15	8.5E-15	8.5E-15	8.5E-15	8.5E-15	4.5E-09
TRANS-8	2.0E-10	2.4E-08	7.8E-09	2.6E-10	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.1E-12	3.2E-08
TRANS-9	1.9E-13	5.2E-11	2.1E-11	7.7E-13	9.5E-15	9.5E-15	9.5E-15	9.5E-15	9.5E-15	9.5E-15	9.5E-15	9.5E-15	9.5E-15	9.5E-15	7.4E-11
	6.4E-07	3.0E-06	5.7E-06	6.0E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	3.9E-06	2.18E-05

Table 4.40

Dominant Component Contributions to Mean Core Damage
Frequency Ranked by Risk Reduction Potential

<u>Component</u>	<u>Percent Reduction if not Failed</u>	
	<u>Base Case</u>	<u>109-NT Upgrade</u>
Building 109-NT	56%	6%
Concrete Silo	10%	32%
Core Inlet Piping	3%	13%
Shield Wall	1%	3%
Building 182-N	1%	2%
Ceramic Insulators	5%	11%
Batteries	0%	0%
Lg. MOV	0%	0%
Sm. MOV and AOV	0%	0%
Lg. Pneum/Hyd Valve	0%	0%
Lg. Manual, Check, Relief Valve	0%	0%
Building 105-N	0%	1%
Building 109-N	0%	0%
Building 181-N	0%	0%
Demineralized Water Tank	0%	0%
13.8 KV Bus	1%	1%
4 KV Bus	1%	1%

4.6.6.2 Basic Event Importance to Overall Uncertainty

The relative contribution of the hazard curve, the seismic response and the seismic fragility uncertainties (β_u 's) to the overall core damage frequency was ascertained. The results of this comparison are shown on Table 4.41. The base case mean, 95 percent and 50 percent core damage frequencies are shown in the first column. The second column shows the corresponding values with the hazard curve fixed at its median value (i.e., with no modeling uncertainty). From the values listed it can be seen that the error factor (EF) associated with these results is 4 whereas the corresponding error factor for the base case was 39. The third column shows a case wherein all the fragility and response modeling uncertainties are simultaneously set to zero. The error factor for this case is 32 which shows that the reduction in response and fragility uncertainties has little effect on the overall core damage uncertainty. These results show quite clearly that the uncertainty in the hazard curve is the dominant factor in both the mean value of core damage frequency and in the uncertainty of the core damage frequency.

4.7 Conclusions

An important assumption implicit in the analyses above is that station blackout (loss of all emergency AC power and battery depletion) does not lead directly to core damage. Given that this is true, then this aspect of the N-Reactor design is a superior design feature not present in the current generation of commercial nuclear power plants. However, if there were subtle dependencies (of which the N-Reactor personnel who performed the Level 1 internal events PRA were not aware) which would invalidate this assumption, then the seismic risk could be significantly higher. The importance of this question is highlighted by the fact that there are a number of (unanchored) transformer common mode failures and block wall buildings with relatively small margin (buildings 151, 153, and 184) whose failure could lead to loss of all onsite AC power. As part of the restart program, it is recommended that this issue be carefully reexamined and the assumption verified.

Overall, the seismic risk assessment has shown that the total core damage frequency ($4.6E-5$ per year for the current configuration or $2.2E-5$ per year for the upgraded building 109-NT case) is comparable to or less than the seismic core damage frequency for two other (more modern) commercial power plants whose risk assessments were performed by the same methods and at the same level of detail. In fact, if the building 109-NT upgrades are performed (to the level assumed in this report), the N-Reactor core damage frequency will be at the low end of the spectrum computed for existing commercial power plants.

Table 4.41

Comparison of Contributions of Modeling
Uncertainty in Response, Fragility and Hazard
Curves to Base Case Core Damage Frequency

	<u>Base Case</u>	<u>Hazard $\beta_U=0$</u>	<u>$\beta_{RR}=0$ $\beta_{FR}=0$</u>
Mean	4.62E-5	5.61E-6	3.27E-5
95%	1.66E-4	1.49E-5	1.18E-4
50%	4.24E-6	4.18E-6	3.71E-6
$\frac{P_{cm}(95\%)}{P_{cm}(50\%)}$	39	4	32

4.8 References

1. E. G. Erpenbeck et al, "N Reactor Seismic Task Force Analysis and Fix Summary," Westinghouse-Hanford Co. Report, WHC-SP-0471, Rev 1, 1990.
2. M. P. Bohn et al, Analysis of Core Damage Frequency: Surry Power Station External Events, NUREG/CR-4550, Vol 3 (Revision 1) Part 3, Nov 1989 (in USNRC Public Document Room).
3. J. A. Lambright, et al, Analysis of Core Damage Frequency: Peach Bottom Unit 2 External Events, NUREG/CR-4550, Vol 4 (Revision 1) Part 3, Oct 1989 (in USNRC Public Document Room).
4. D. L. Bernreuter, et al., Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains, Vols I-III, NUREG/CR-5250, October 1988.
5. Electric Power Research Institute, Seismic Hazard Methodology for the Central and Eastern United States, EPRI NP-4726, Vols 1-10, July 1986
6. D. L. Bernreuter, Seismic Hazard Analysis, Vols 1-4, NUREG/CR-1582, October 1981
7. R. B. Kulkarni, et al., Evaluation of Seismic Exposure for the N-Reactor Plant, Hanford, WA, Report prepared by Woodward-Clyde Consultants, published as UNC Nuclear Industries, Inc. report UNI 4426, May 1987.
8. Washington Public Power Supply System. Final Safety Analysis Report, WPPSS Nuclear Project No. 2, Amendments 18, 21, and 36, Washington Public Power Supply System, Richland, Washington, 1981..
9. R. P. Kennedy, et al., Engineering Characterization of Ground Motion. Vol 1, NUREG/CR-3805, May 1984.
10. O. R. Maslenikov, et al., Soil Structure Interaction analysis and In-Structure Response Spectra Generation for the N-Reactor Facility, prepared for UNC Nuclear Industries by EQE Engineering, Inc., Vols. I and II, May 1987.
11. P. S. Hashimoto, et al., N-Reactor River Pump House and Gantry Crane (181-N) Seismic and Tornado Analysis, prepared for Westinghouse Hanford Company by EQE Engineering, Inc., March 1988.
12. E. O. Weiner, "N Reactor Seismic Probabilistic Risk Assessment-182N and 109N Turbine Extension Building Analyses", Westinghouse - Hanford Co. Report SD-NR-SA-019, June 1989.

13. P. B. Schnabel, J. Lysmer, and H. B. Seed, SHAKE--A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites, Earthquake Engineering Research Center, University of California, Berkeley, CA, EERC 72-12, 1972.
14. N-Reactor Updated Safety Analysis Report, United Nuclear Industries, Richland, Washington, UNI-M-90, Vol. 2, Rev. 9-14-84.
15. J. C. Chen, et al., Seismic Responses of N-Reactor Core, Independent Review of Phase II Work, prepared by Lawrence Livermore National Laboratory for UNC Nuclear Industries, UNI-3179, August 1985.
16. H. B. Seed, C. Ugas, and J. Lysmer, Site Dependent Spectra for Earthquake Resistant Design, Earthquake Engineering Research Center, University of California, Berkeley, CA, EERC 74-12m, 1974.
17. J. J. Johnson, G. L. Goudreau, S. W. Bumpus, and O. R. Maslenikov, Seismic Safety Margins Research Program Phase I Final Report--SMACS - Seismic Methodology Analysis Chain with Statistics (Project VIII), Lawrence Livermore National Laboratory, Livermore, CA, UCRL-53021, Vol. 9, NUREG/CR-2015, Vol. 9, 1981.
18. H. L. Wong and J. E. Luco, Soil-Structure Interaction: A Linear Continuum Mechanics Approach (CLASSI), Dept. of Civil Engineering, University of Southern California, Los Angeles, CA, CE79-03, 1980.
19. J. Lysmer, et al., FLUSH - A Computer Program for Approximate 3-D Analysis of Soil-Structure Interaction Problems, Earthquake Engineering Research Center, University of California, Berkeley, EERC 75-30, 1975.
20. Seismic and Pressure Tube Rupture Evaluation of DOE N-Reactor -Phase II Report, Vol. 2, prepared by Impell Corporation for UNC Nuclear Industries, CA, 1985.
21. P. H. Estes, Seismic Analysis of 60 Ton Gantry Crane, UNC Nuclear Industries, UNI-2783, March 1984.
22. M. P. Bohn et al., Application of the SSMRP Methodology to the Seismic Risk at the Zion Nuclear Power Plant, Lawrence Livermore National Laboratory, Livermore, CA, UCRL-53483, NUREG/CR-3428, 1983.
23. Memo from E. O. Weiner (Westinghouse Hanford Company) to M. J. Mraz (EQE), December 5, 1988.

24. National Academy of Science, Safety Issues at the Defense Production Reactors - A Report to the Department of Energy, National Academy Press, 1987.
25. B. G. Rabbat and H. G. Russell, "Friction Coefficient of Steel on Concrete or Grout," Journal of the Structural Division, ASCE (1985).
26. U.S. Atomic Energy Commission, Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973.
27. J. Lysmer and J. M. Duncan, Stresses and Deflections in Foundation and Pavements, fourth edition, U.C. Berkeley Department of Civil Engineering, Institute of Transportation and Traffic Engineering, 1969, pp. 1-18.
28. Gregory Szuladzinski, Dynamics of Structures and Machinery - Problems and Solutions, The Wiley Co., New York, 1982.
29. Electric Power Research Institute, Seismic Ruggedness of Aged Electrical Components, EPRI NP-5024, January 1987.
30. L. E. Cover, et al., Handbook of Nuclear Power Plant Seismic Fragilities, NUREG/CR-3558, December 1983.
31. R. D. Campbell, et al., Compilation of Fragility Information From Available Probabilistic Risk Assessments, Lawrence Livermore National Laboratory Report UCID-20571 Rev. 1, September 1988.
32. D. A. Wesley, and P. S. Hashimoto, Seismic Structural Fragility Investigation for the Zion Nuclear Power Plant, NUREG/CR-2320, October 1981.
33. R. Riddell and N. M. Newmark, Statistical Analysis of the Response of Nonlinear Systems Subjected to Earthquakes, University of Illinois, UILU-ENG 79-2016, August 1979.
34. R. P. Kennedy, et al., Engineering Characterization of Ground Motion - Task I: Effects of Characteristics of Free-Field Motion on Structural Response, NUREG/CR-3805, May 1984.
35. R. Meli, "Behavior of Masonry Walls Under Lateral Loads," presented at the Fifth World Conference on Earthquake Engineering, Rome, Italy, 1972.
36. M. J. N. Priestley, and D. O. Bridgeman, "Seismic Resistance of Brick Masonry Walls," Bulletin of the New Zealand National Society For Earthquake Engineering (December 1974).

37. M. J. N. Priestley, "Seismic Resistance of Reinforced Concrete Masonry Shear Walls With High Steel Percentages," Bulletin of the New Zealand National Society For Earthquake Engineering (March 1977).
38. M. J. N. Priestley and D.M. Elder, "Cyclic Loading Tests of Slender Concrete Masonry Shear Walls," Bulletin of the New Zealand National Society For Earthquake Engineering (March 1982).
39. A. E. Cardenas, et al., "Design Provisions for Shear Walls," Journal of the American Concrete Institute (March 1973).
40. R. R. Wyer, Safe Shutdown Earthquake and Design Basis Tornado Evaluation for the Emergency Raw Water Bypass Silo, U-12013, Revision 0, July 1987.
41. J. A. Blume, N. M. Newmark, and L. H. Corning, Design of Multistory Reinforced Concrete Buildings for Earthquake Motions, Portland Cement Association, 1961.
42. D. A. Wesley, R. P. Kennedy, and P. J. Richter, Analysis of the Seismic Collapse Capacity of Unreinforced Masonry Wall Structures, presented at the Seventh World Conference on Earthquake Engineering, Istanbul, Turkey, September 8-13, 1980.
43. R. R. Wyer, Safe Shutdown Earthquake and Design Basis Tornado Evaluation for the Demineralized Water Tank, prepared for UNC Nuclear Industries by Westinghouse Hanford Company, UNI-4461, Revision 0, June 1987.
44. S. N. Huang, Seismic Fragility Study of the N Reactor Emergency Raw Water Tank, Westinghouse - Hanford Report SD-NR-SA-015, November 1988.
45. C. B. Orpeda, N Reactor PRA Seismic Piping Analysis, Westinghouse - Hanford Co. Report SD-NR-SA-022/Revision 0, August 1989.
46. R. B. Worrell, SETS Reference Manual, NUREG/CR-4213, May 1985.

5.0 N-REACTOR FIRE ANALYSIS

5.1 Summary

A detailed fire risk assessment has been completed for the N-Reactor to estimate the contribution of fire-induced events to core damage frequency. The NUREG-1150 external events methodology used for this analysis consists of two parts. First, a preliminary screening analysis is conducted to identify plant areas with the potential to contribute to core damage. Then more detailed analyses are made for these areas to quantify core damage frequencies.

Detailed screening analyses found that most plant areas had a negligible contribution to fire-induced core damage frequency. The following fifteen areas had core damage contributions of greater than $1.0E-8$ per year.

- a. Cell Access Corridor
- b. Building 153 Main Floor "B" Switchgear
- c. Building 153 Main Floor "A" Switchgear
- d. Building 184 Instrument Air Compressor Cable Runs
- e. Building 184 Compressor Area (Instrument Air)
- f. Building 182 Control Room
- g. Building 105, Room 9
- h. Building 105, Room 8
- i. Outlet Piping Area
- j. Building 105, Room 6 (Cable Spreading Room)
- k. Building 105, Corridor 1
- l. Building 109, Corridor 19 and Building 109-NA Basement
- m. Main Control Room
- n. Building 151 Switchgear
- o. Building 184 Control Room

The overall total (mean) core damage frequency was found to be $2.0E-06$ per year. Table 5.1 lists the mean core damage frequency contributions for each of the ten most important fire scenarios in descending order of importance. The accident sequences these scenarios mapped into are listed in Table 5.2.

5.1.1 Remedial Engineering Measures

During conduct of the fire PRA, detailed tracing of control and power cabling revealed that fires caused by ignition of transient combustible materials could damage most plant safety functions in cable "pinchpoint" areas. Three plant areas were identified early in the analysis as potential dominant contributors to fire-induced core damage. These are the first three fire areas listed in Table 5.1 including:

- a. Corridor 19 and Building 109-N Basement
- b. Building 105 Corridor 1
- c. Main Control Room

Table 5.1

Dominant N-Reactor Fire Area Core Damage Frequency Contributors (/yr)

<u>Fire Area</u>	<u>Base Case</u>	<u>Sensitivity Study</u> <u>(plant prior to fixes)</u>
Corridor 19 and Building 109 Basement	9.7E-6	1.3E-4
Building 105, Corridor 1	1.8E-7	4.6E-5
Main Control Room	6.5E-6	3.2E-5
Building 184 Instrument Air Compressor Cable Runs	1.3E-6	N/A
Building 151 Switchgear	7.0E-7	N/A
Building 105 Room 6	6.6E-7	N/A
Building 105 Room 8	1.6E-7	N/A
Building 105 Room 9	1.6E-7	N/A
Building 153 Main Floor "B" Switchgear	1.5E-7	N/A
Building 153 Main Floor "A" Switchgear	7.0E-8	N/A
<u>Total</u>	<u>2.0E-5</u>	<u>2.1E-4</u>

Base case results listed for these three fire areas in Table 5.1 are dependent on completion of fire protection engineering changes in these areas prior to N-Reactor re-start.

The current plant configuration has limited sprinkler coverage in Corridors 1 and 19. As a sensitivity study the current sprinkler configurations for these areas were analyzed for each fire scenario. Corridor 1 fire risk is increased by approximately a factor of twenty if fire protection systems are left unchanged as shown in the Table 5.1 column for "sensitivity studies." A similar study for Corridor 19 indicates that fire risk increases by a factor of approximately ten. Thus, sprinkler protection and fire alarm upgrades in these two areas are important improvements necessary to limit the facility fire risk.

Table 5.2

Dominant Accident Sequence Core Damage Frequency Contributors

<u>Sequence</u>	<u>Fire Area</u>	<u>Mean Core Damage Frequency (/yr)</u>
T9	Corridor 19, Bldg. 109 Basement	9.7E-6
	Main Control Room	6.5E-6
	Corridor 1	1.8E-7
	Room 6	6.5E-7
T8	Room 9	1.5E-7
	Room 8	1.6E-7
	Outlet Piping Area	1.6E-8
T4	Bldg. 184 Cable Runs	1.3E-6
	I.A. Compressor Area	3.2E-8
	Bldg. 153 "B" Switchgear (Main Floor)	1.4E-7
	"A" Switchgear (Main Floor)	6.0E-8
	Bldg. 151 Switchgear	8.0E-7
	Control Room Bldg 182	2.8E-8
	Cell Access Corridor	2.2E-8
	Bldg. 184 Control Room	2.1E-8

Other upgrades are needed to improve the fire risk for the main control room. Existing procedures do not adequately address the question of control room abandonment during an uncontrolled fire. These procedures need to be improved to provide for orderly transfer of operator control from the main control room to an existing alternate shut-down panel. Acceptance testing of the alternate shut-down panel has not yet been completed. Both the procedure upgrades and acceptance testing should be completed prior to N-Reactor re-start. The effect of these upgrades on Main Control Room fire risk is also shown in Table 5.1 as a sensitivity study.

Comparison of the total N-Reactor fire core damage frequencies listed in Table 5.1 under "base case" and "sensitivity studies" indicates that relatively simple, low-cost facility modifications can have a major impact on the overall fire risk of operating N-Reactor.

5.1.2 Comparison to Operating Commercial Reactors

The fire risk assessment methodology used for the N-Reactor analysis is still under development. This method has only been applied to two commercial nuclear power plants which were part of the NUREG-1150 program. These plants are Surry Unit 1, a Westinghouse three-loop PWR which came on line in 1972, and Peach Bottom Unit 2, a General Electric BWR-4 reactor which came on line in 1974. Both of these commercial units have completed Appendix R fire protection retrofits as part of their ongoing operation.

Figure 5.1 provides a comparison between the fire risk at N-Reactor for the "base case" reported above and the Surry and Peach Bottom commercial power reactors. This figure illustrates that N-Reactor fire risk is comparable to that of two somewhat more recent commercial power reactors both of which have received Appendix R 10 CFR 50 retrofits. N-Reactor has not been brought into compliance with Appendix R. Remedial fire protection measures were identified during the conduct of fire PRAs at all three facilities, and have been completed in a similar fashion (engineering modifications at N-Reactor are, however, contingent on a re-start order).

5.1.3 Methodology Overview

5.1.3.1 Initial Plant Visit

Based on the internal event and seismic analyses, the general location of cables and components of the systems of interest is known. The plant visit provides the analyst with a means of seeing the physical arrangements in each of these areas. The analyst will have a fire area checklist which will aid the screening analysis and in the quantification

Figure 5-1. Comparison of N Reactor Fire Risk to NUREG 1150 Commercial Nuclear Power Plants.

step. The second purpose of the initial plant visit is to confirm with plant personnel that the documentation being used is, in fact, the best available information and to get clarification about any questions that might have arisen in a review of the documentation.

Also, a thorough review of firefighting procedures is conducted. This review was conducted to determine probability of manual suppression in a given length of the time for all critical plant areas. The results of this analysis are given in Appendix A.

5.1.3.2 Screening

It is necessary to select important fire locations within the power plant under investigation having the greatest potential for producing risk-dominant accident sequences. The objectives of location selection are somewhat competing and should be balanced in a meaningful risk assessment study. The first objective is to maximize the possibility that all important locations are analyzed, and this leads to the consideration of a potentially large number of candidate locations. The second objective is to minimize the effort spent in the quantification of event trees and fault trees for fire locations that turn out to be unimportant. A proper balance of these objectives is one that results in an ideal allocation of resources and efficiency of assessment.

The screening analysis is comprised of:

- a. Identification of relevant fire areas. Fire areas which had either safety-related equipment or power and control cables for that equipment were identified as requiring further analysis. This group of fire areas (areas) is briefly described in Section 5.2. All critical safety components within these fire areas are given in Appendix B.
- b. Screen fire areas on probable fire-induced initiating events. Determination of the fire frequency for all plant locations and determination of the resulting fire-induced initiating events and "off-normal" plant states is delineated in Sections 5.3 and 5.4, respectively.
- c. Screen fire areas on both order and frequency of cut sets.
- d. Each fire area remaining is numerically evaluated and culled on probability.

The screening methodology (Section 5.5) describes how reduction of the initial group of fire locations from Section 5.2 to the fifteen remaining with contributions to core damage frequency of greater than 10^{-8} per year was accomplished.

5.1.3.3 Quantification

After the screening analysis has eliminated all but the probabilistically-significant fire areas, quantification of dominant cut sets is completed as follows:

- a. Determine temperature response in each fire area.
- b. Compute component fire fragilities. The latest version of the fire growth code COMPBRN with some modifications was used to calculate fire propagation and equipment damage. A description of these results for steps 1 and 2 is given in Section 5.6. These fire calculations were only performed for the fire areas that survived the screening analysis.
- c. Assess the probability of barrier failure for all remaining combinations of fire areas. A barrier failure analysis was conducted for those combinations of two adjacent fire areas which, with or without additional random failures, remained after the screening analysis. The methodology to assign barrier failure probability to the fire area combinations is described in Section 5.7.
- d. Perform a recovery analysis. In similar fashion as in the internal event analysis, recovery of non-fire related random failures was addressed. Appropriate modifications to recovery probabilities were made as described in Section 5.8.
- e. An uncertainty analysis is performed to estimate error bounds on the computed fire-induced core damage frequencies. The TEMAC code was utilized in the uncertainty analysis as described in Section 5.9.

In Section 5.10 a detailed description of all fire scenarios with contributions to core damage frequency of greater than 10^{-8} per year and their associated fire areas is given. Distributions and description of all factors used in the final quantification of all fire areas are delineated.

5.2 Fire Locations Analyzed

In this section, the plant areas are listed and discussed. Table 5.2 provides a comprehensive listing of the fire areas and a brief physical description of each. A list of the components and basic events

Table 5.3

N-Reactor Fire Area Descriptions

Fire Area	Building	Elevation	Physical Description
1D	105N	-15 Ft	Rooms 10 and 13, Outlet Piping Space; "D" Elevator; Stairways 9, 39, and 40.
3A	105N	-15 Ft	Room 6, Cable Spreading Room.
3B	105N	-15 Ft	Corridor 2; Rooms 2, 7, 8, 9, and 12; Electrical Tunnel (Room 47); Electrical Cable Chase Above Zone I Supply Plenum; Stairway 25.
3B1	105N	-15 Ft	Corridor 1.
3G	105N	-15 Ft	Rooms 29 and 30; "W" Elevator Shaft; Stairway 3.
3J	105N	-15 Ft	Room 39; Corridors 4 and 5; Airlocks 25 and 38.
4A	105N	0 Ft	Main Control Room (145); Rooms 146, 147, 148, and 149; Corridor 8B.
21	109N	-16 Ft	Rooms 9 (Turbine Drive Bay Basement), 12, 13, 14, 15, 16, 17, 20, 21, 22, 23, 24, 25, 30, 31, 32, 33, 34, 35; Corridor 19; Airlock 18; Stairways 1, 4, 5, 6.
		0 Ft	Rooms 107 (Turbine Drive Bay), 101, 102, 103, 104, 105, 106, 108, 110, 111, 112, 113, 114, 115, 116, 118, 120; Corridor 109; Stairways 2, 4A, and 7.
		+12 Ft	Rooms 207, 208, 209.
		+24 Ft	Rooms 301, 303, 304, 305, 306, 307, 308.
22	109N	-16 Ft	Rooms 10 and 37 (Cell Access Corridor); Airlocks 11 and 36.

Table 5.3

N-Reactor Fire Area Descriptions (Concluded)

<u>Fire Area</u>	<u>Building</u>	<u>Elevation</u>	<u>Physical Description</u>
25A	151N	-14 Ft	Bldg. 151 Basement.
25B	151N	0 Ft	Bldg. 151 Main Floor.
26A	153N	-13 Ft	Bldg. 153 Basement A Bus Side.
26B	153N	-13 Ft	Bldg. 153 Basement B Bus Side.
27A	153N	0 Ft	Bldg. 153 Main Floor A Bus Side.
27B	153N	0 Ft	Bldg. 153 Main Floor B Bus Side.
30A	181N	421' -0"	Bldg. 181 Main Pump House.
30B	181N	421' -0"	Bldg. 181 Annex.
30C	181N	421' -0"	Bldg. 181 Outside Area Around CRW Pumps.
31A	182N	-24' -6"	Bldg. 182 Basement.
31B	182N	0 Ft	Bldg. 182 Main Floor; Rooms 101, 102, 104, 105, 106, and 107.
31C	182N	0 Ft	Bldg. 182 Control Room.
32A	184N	0 Ft	Bldg. 184 All Areas, All Elevations.
32B	184N	0 Ft	Bldg. 184 Annex.

associated with each of the critical fire areas is provided in Appendix B. The tables contain brief discussions of the fire areas which were analyzed. All other fire areas not contained in these tables were screened as they did not contain either vital equipment or cabling for such equipment was not routed through the areas.

Determination of fire areas and the boundaries or barriers between respective areas was made based on a comprehensive analysis of the plant layout drawings, supplemented with plant walkdowns to verify the selections made and to make modifications in those cases where adequate information was not available on the drawings. Fire area determinations were made along major plant functional area boundaries based on the existing divisions from the general arrangement drawings. Some divisions were based on the existing "zones" used to presently divide the plant up for ventilation and contamination control.

Without the benefit of an Appendix R-type submittal characteristic of commercial nuclear facilities, along with significant differences in the design and layout of plant structures, cabling and equipment, it was necessary to take a somewhat different approach to the designation of fire areas than is usual for commercial nuclear plants. Typically, fire areas are bounded by 3-hr rated fire barriers; fire doors, fire-resistant walls, fire dampers, and the like. These barriers are not often present at N-Reactor. Therefore, it was necessary to make somewhat arbitrary divisions of some of the areas into discrete zones isolated by non-rated barriers such as shield walls and normally closed doors. Some of the areas were selected based on the need to limit the size of the fire areas to allow meaningful assessment of fire frequency and to better analyze manual suppression probabilities.

The subdivision into fire areas was also constrained by building elevation. When stairwell doorways were determined to be normally closed, despite the lack of 3-hr fire ratings, it was deemed prudent to count the areas adjacent to the stairwell as independent fire areas. In the cases where stairways were not isolated by doors or open gratings were present, the rooms at the different elevations were considered part of a single fire area.

5.3 Initiating Event Frequencies

Data on fires in Light Water Reactors have been analyzed in several studies (Ref. 1,2,3). Although they have been done independently, they have some common aspects. For example, almost all studies have used License Event Report (LER) data from the Nuclear Regulatory Commission (NRC). All have reported the overall frequency of fires of approximately 0.16 per reactor year on a plant wide basis. Since N-Reactor fire occurrence frequency is similar to this generic data, plant-specific frequencies were developed as is explained in the following paragraphs.

To determine fire initiating event frequencies, there are two kinds of information needed: (1) the number of fire incidents that have occurred in specific compartments during commercial operation, and (2) the number of compartment years that the nuclear industry has accumulated. Most of the data for the first part comes from reports of insurance inspectors to American Nuclear Insurers (ANI), although other sources are also used, e.g., the U.S. Nuclear Regulatory Commission. While the NRC requires the reporting of fires that, in some way, affect the safety of the plant, the ANI has more stringent requirements in the sense that all fire events must be reported. Compartment years are computed by adding the age of all compartments (within a certain category of compartments) of units that were in commercial operation by the end of June 1985. The age is defined as the time between first commercial operation and the end of June 1985 (or date of decommissioning). The combination of specific fire locations and compartment age is given in Table 5.4. Even though fire events that occurred when the plant was shutdown were used, an event was only included if it could be postulated that it also might occur when the plant was at power. Eight areas are typically found in nuclear power plants. These are (1) the control room, (2) cable spreading room, (3) diesel generator room, (4) reactor building, (5) turbine building, (6) auxiliary building, (7) electrical switchgear room, and (8) battery room. In most plants, the first three areas and the electrical switchgear room and battery room are single compartments while the other three are typically large buildings. A detailed listing of the generic fire events can be found in Appendix C.

Table 5.4

Statistical Evidence of Fires in LWRs
(As of June 1985)

Area	Number of Fires r	Number of Compartment Years T
Control Room	3	681.0
Cable Spreading Room	2	747.3
Diesel Generator Room	37	1600.0
Reactor Building	15	847.5
Turbine Building	21	654.2
Auxiliary Building	43	673.2
Electrical Switchgear Room	4	1346.4
Battery Room	4	1346.4

To obtain fire area-specific initiating frequencies, a partitioning method is required. Partitioning allows the analyst to subdivide the frequency of fire occurrence from a large building (e.g., auxiliary building) to a specific room or area within that building. Also, further partitioning can occur within a specific room or area. One method of partitioning is comprised of ratioing the floor areas of fire areas within a building. The assumption here is that the probability of fire occurrence is dependent only upon the amount of floor area a particular fire area contains. Another method of partitioning would look at each fire area and analyze factors important to probability of fire initiation. These factors are the amount of electrical components and cabling, the fire loading, whether the fire area is controlled, and how often the fire area is occupied.

The fire events and operating years for the eight plant areas were obtained using the fire data base developed by Wheelis (Ref. 4). To determine operating years for electrical switchgear rooms and battery rooms, auxiliary building operating years were doubled. A survey of all U.S. light water reactors indicated that there is an average of 2.25 trains of emergency switchgear and their associated batteries per plant. However, it is known that some plants such as Surry locate both trains of their emergency switchgear in one fire area. So it was assumed that an average number would be close to two per plant for both types of rooms.

To aid partitioning within a large building or within a specific fire area in that building a checklist was used on the initial plant visit to determine the most probable fire initiating sources. Also, data on past fire occurrences was thoroughly reviewed. For instance, control room data indicate that fires have only occurred in electrical cabinets. Therefore, area ratios were developed based on cabinet area within this respective area. Since a transient combustile-initiated fire has never occurred in a control room, this type of fire scenario was eliminated from further consideration for the control rooms in Buildings 105, 182, and 184.

The generic fire occurrence data was updated using a method developed by Iman (Ref. 5) to determine plant specific fire occurrence frequencies.

This Bayesian approach models the incidence rate for each plant relative to the incidence rates of all other plants, and the posterior distribution is found for the incidence rate for each plant.

For this analysis the gamma distribution is used as a model, although many other distributions could be used. The probability density function for the two-parameter gamma distribution is:

$$h(\lambda) = f_{\gamma}(\lambda|\alpha, \beta) = \beta^{\alpha} [\Gamma(\alpha)]^{-1} \lambda^{\alpha-1} e^{-\beta\lambda} \quad \lambda \geq 0, \alpha, \beta > 0$$

The parameters α and β are unknown, and the noninformative prior is:

$$p(\alpha, \beta) = 1/(\alpha\beta) \quad \alpha, \beta > 0$$

The likelihood function of the datum (s_i, t_i) is Poisson

$$L(s_i, t_i | \lambda_i) = (\lambda_i t_i)^{s_i} e^{-\lambda_i t_i} / s_i!$$

The posterior density can, therefore, be expressed as:

$$p^*(\alpha, \beta, \lambda_0, \dots, \lambda_n) =$$

$$\frac{p(\alpha, \beta) \prod_{i=0}^n [h(\lambda_i) L(s_i, t_i | \lambda_i)]}{\int_0^\infty \int_0^\infty \int_0^\infty \dots \int_0^\infty p(\alpha, \beta) \prod_{i=0}^n [h(\lambda_i) L(s_i, t_i | \lambda_i)] d\alpha d\beta d\lambda_0 \dots d\lambda_n}$$

In this way plant specific fire initiating event frequencies and distributions were developed. Table 5.5 lists the N-Reactor specific fire initiating event frequencies for the eight types of fire areas with contributions to core damage frequency of greater than 10^{-8} per year.

Table 5.5

N-Reactor Fire Initiating Event Frequencies (/yr)

<u>Fire Area</u>	<u>Mean</u>	<u>5th Percentile</u>	<u>50th Percentile</u>	<u>95th Percentile</u>
Building. 109	4.3E-2	2.2E-2	3.0E-2	1.1E-1
Bldg. 105	2.0E-2	1.1E-2	1.7E-2	3.3E-2
Electrical Switchgear Room	2.8E-3	6.2E-6	2.6E-3	5.9E-3
Bldg. 184	6.5E-2	3.0E-2	3.2E-2	1.4E-1
Control Room	3.2E-3	1.7E-7	3.0E-3	7.4E-3
Battery Room	2.7E-3	1.6E-7	2.4E-3	6.1E-3
Confinement	5.6E-3	2.1E-4	4.8E-3	1.1E-1
Room 6	7.5E-3	1.1E-3	4.3E-3	2.6E-2

A comprehensive review of all N-Reactor unusual occurrence and event reports was conducted to determine what significant fires had occurred. Table 5.6 lists the fire events used in the initiating event frequency updating procedure.

These events were compared to fire events which were reported in the data base developed by Wheelis (Ref. 4). Any N-Reactor events of a similar nature were included to insure consistency between data. A number of N-Reactor reports were excluded based on the following criteria; (1) no actual fire occurred only the appearance of some smoke, (2) fire alarms or deluge systems tripped without the presence of a fire, and (3) fires that occurred in non-vital areas or during construction of the plant.

5.4 Determination of Fire-Induced "Off-Normal" Plant States

One of the most critical steps in a fire analysis is to determine on a plant specific basis which of a wide range of possible initiating events have the potential to be induced due to a fire occurrence.

As in the N-Reactor internal events analysis, a comprehensive list of initiators was identified for further study. It is known from a review of previous fire PRAs that only a limited set of initiating events have the potential to be significant contributors to fire-induced core damage frequency. Typically, initiating events such as large or medium LOCAs caused directly by the fire have not been analyzed because the vulnerabilities of a piping system or tanks to fire events are considered insignificant.

Table 5.7 lists the initiating events that were retained after the screening process and provides a brief explanation as to why a particular initiating event was included for further study.

A comprehensive look at system drawings was conducted to determine the potential for large or small LOCAs caused by spurious valve actuation. No probabilistically significant mechanism could be found during this review. The frequency of all potential fires leading to a hot short-induced LOCA were judged to be $<1.0E-6/\text{yr}$. These assessments did not take into account the additional random failures required to lead to core damage or operator recovery of such random failure. Such additional failures would substantially reduce any core damage frequency contribution. It is also known from past fires (such as at Brown's Ferry) even if spurious actuations would occur, that within approximately one-half hour spurious actuations will cease due to hot shorts terminating in open circuits. Therefore, large and small LOCAs were eliminated from further consideration.

The same fault trees and event trees that were used in the internal events analysis were utilized in the fire analysis. Thus, the level of analytical detail was consistent with the level in the internal event analysis.

5.5 Detailed Description of the Screening Analysis

A comprehensive screening analysis is required to reduce the number of potential fire-induced scenarios to only those which have the potential to be probabilistically significant to core damage frequency.

Table 5.6

N-Reactor Fire Occurrences

Date	Plant Status	Bldg.	Location	Fire Type	Description
12/08/72	Shutdown	109	MCC #20 for #10 Exh Fan	Electrical	Loose connection and arcing caused cubicle fire in MCC #20 when attempting to start #10 exhaust fan. Suppressed with hand-held extinguisher.
06/12/73	Extended Outage	105	Cable Spreading Room	Electrical-Relay Panel	Room 6 relay overheated and ignited in relay panel PR-39, with smoke and fire damage to other relays and wiring. Used portable CO ₂ extinguisher; no propagation beyond panel.
10/20/74	Operation	181	CRW Pump Motor	Electrical	Fire in #4 CRW pump motor; exciter brushes arced and ignited. Portable fire extinguisher used to suppress.
11/11/74	Shutdown	184	CE Boiler	Fuel Oil	Burner valve not closed; oil leaked into CE boiler fire box and ignited. Fire burned itself out.
04/25/75	Operation	105	Control Rod Pump Bearing	Bearing Lube Oil	A small fire due to an overheated bearing failed a control rod recirc pump (#3). Extinguished with chemical agent "quickly."
09/24/78	Operation	109	MCC #20 and Zone I Exh Fan	Electrical	Zone I exhaust fan #4 motor seized and burned and the breaker in MCC #20 burned due to motor bearing failure.
05/28/79	Cold Shutdown	182	EDG Engine Fire	Oil - Diesel	Emergency diesel engine fire due to oil leak which was ignited by fuel injector solenoid. Suppressed manually with portable CO ₂ extinguisher.

Table 5.6

N-Reactor Fire Occurrences (Concluded)

<u>Date</u>	<u>Plant Status</u>	<u>Bldg.</u>	<u>Location</u>	<u>Fire Type</u>	<u>Description</u>
04/08/80	Operation	109	Roof Fire	Oil	Residual oil and grease on 109 roof ignited by hot cutting torch slag and extinguished by portable means.
11/09/83	Operation	109	Cell No. 6	Oil-Soaked Fire Hose	Welding in cell no. 6 ignited a canvas fire hose in the -16 ft C sump, into which primary pump oil had flowed. Fire department extinguished.

Table 5.7

N-Reactor Fire-Induced Initiating Events Analyzed

Initiating Event	Comments
Transient	Similar to the seismic methodology if no other initiator could occur it was assumed that either the operator would manually shutdown the reactor or that fire-related damage would cause an automatic reactor trip.

The screening analysis is composed of the following five steps:

Step 1. Identification of Relevant Fire Areas

Fire areas containing equipment or cables associated with safety-related systems which mitigate the effects of the unscreened fire-induced "off-normal" plant states were identified. All other fire areas were then eliminated from further analysis. This resulted in the fire areas which are described in Section 5.2.

Step 2. Screen Fire Areas on Frequency of Significant Fire

Fire initiating event frequencies were developed utilizing the data base developed by Wheelis (Ref. 4) and updated to N-Reactor plant-specific frequencies using the methodology developed by Iman (Ref. 5). If it could be shown that the frequency of a fire alone for a given fire area was less than 10^{-6} per year, then that fire area was eliminated from further consideration.

Step 3. Screen Fire Areas Based on Vital Area Analysis

The remaining fire areas underwent a vital area analysis (location mapping) of components as well as control and power cables for a limited set of "vital" components that were located within these areas. (Appendix B contains a listing of components within the remaining fire areas.) This information resulted in a transformation block used in conjunction with the SETS computer code (Refs. 6,7) to solve all front line systems and then solve all of the identified sequences (Table 5.7) of Section 5.4.

Fire occurrence frequency for each fire area was set to 1.0 and, given a fire, all components within that area were assumed to fail. The output of this process was accident cut sets which had fire area combinations as well as random failures (i.e., not fire-related) included.

Truncation of cut sets at a random failure probability of 10^{-4} was accomplished which is equivalent to truncation of internal event cut sets at approximately 10^{-8} since the fire frequency is arbitrarily set for screening purposes to 1.0 (but in actuality is approximately 10^{-4}).

Cut sets which contained two fire areas were screened on the following three criteria: (1) no adjacency between areas, (2) no penetrations in the adjacency between areas, and (3) if there were penetrations by numerical culling with barrier penetration failure set to a screening value of 0.1. It is known from the analysis of many fire barriers that typical failure rates are on the order of 10^{-2} to 10^{-3} (Ref. 14). Therefore, this screening value has been set high enough to insure potentially important fire area combinations are not lost.

One additional important piece of information gained from these cut sets was identification of the remaining plant locations where area-to-area barriers needed to be analyzed. Dominant cut sets which contained adjacent fire areas were analyzed for barrier failure in the quantification process.

Step 4. Cull Fire Areas on Probability

Cut sets not eliminated in the first three screening steps were resolved with fire area-specific initiating event frequencies that were calculated in Section 5.3.

Also, operator recovery of non-fire related random failures was included. For screening purposes all short term (less than 24 hr) recovery action failure probabilities (of non-fire failures) were increased from their respective internal events probabilities by a factor of five to allow for the additional confusion of the fire situation occurring in conjunction with other random failures. If recovery actions were long term (greater than 24 hr) no modification to internal event probabilities was deemed appropriate. It is felt that by this time the fire will be extinguished and any spurious signals will have terminated in open circuits.

Step 5. Confirmatory Plant Visit

For those remaining fire areas all fire-related failure scenarios were identified. A scenario can be thought of as a combination of one or more fire related equipment failures within a fire area with or without additional non-fire related (random) failures outside of the fire area. These failure combinations must minimally lead to core damage. Each fire area can have one or more scenarios depending on the equipment combinations which must fail due to the fire in that particular area. A second plant visit was then conducted to determine which of these scenarios were valid based upon cable or equipment locations within a particular fire area. For instance, if a given scenario required the fire-related failure of cabling for components A and B and it could be

shown that these cables were always separated by greater than 40 ft or one of the component's cabling was in a 3-hr rated fire wrap, then these types of scenarios were eliminated from further consideration. Past experience with fire code calculations, which is discussed in the following section, and fire testing, provided much of the basis for assessing the validity of the scenarios.

5.6 Fire Propagation Modeling

The COMPBRN fire growth code (Ref. 8) was used to calculate fire propagation and equipment damage. COMPBRN was developed specifically for use in nuclear power plant fire PRAs. The code calculates the time to damage critical equipment given that a fire has started. This failure time is then used in conjunction with plant-specific information on fire suppression to obtain the probability that a given fire will cause equipment failure which leads to core damage before the fire can be suppressed. The latest version of the code, COMPBRN III (Ref. 9) with some additional modifications was used for the calculations.

COMPBRN follows a quasi-static approach to simulate the process of fire during the pre-flashover period in an enclosure. COMPBRN uses a zone model breaking the fire environment into three zones: flame/plume, hot gas layer, and ambient (see Figure 5.2). Simple fire and heat transfer models and correlations are employed to predict the thermal environment as a function of time. The thermal response of various targets in the fire scenario is modeled to predict the amount of time required for a fire to damage or ignite critical equipment. The critical equipment is generally taken to be a cable tray carrying cables necessary for safe shutdown of the plant, although other critical components such as pumps may be modeled.

The original version of COMPBRN, now referred to as COMPBRN I, has been used to calculate damage times in the majority of fire PRAs to date. However, the code calculations are thought to be highly conservative due to the neglect of heat losses from the targets. A critical assessment of the code containing this and other problems has been performed (Ref. 10). In response to these problems with COMPBRN I, two later versions of the code were developed: COMPBRN II and COMPBRN III (Ref. 9). Neither of the later versions of the code have been extensively validated or compared to data, but presumably represent various degrees of improvement.

As a part of a recent study (Ref. 3) on nuclear power plant fire risk assessment, the latest version of the code (COMPBRN III) was selected to requantify fire damage times from several fire PRAs. Initial attempts to use COMPBRN III in the requantification resulted in the observation of problems with and nonphysical behavior of the code. Many of the code calculations could not be explained on a physical basis. As a result of the observed nonphysical behavior of the code, an effort was undertaken to identify problem areas and to suggest and implement modifications to the code which make the code predictions more reasonable on a physical basis. It was this modified version of the COMPBRN code which was used

Figure 5.2 COMPBRN Zone Model

to provide the fire propagation analysis for this report. References 3 and 11 provide detailed discussions of the problems noted and recommended modifications for the COMPBRN III code. The following is a brief listing of the major problems which were identified and addressed in the modified version of the code:

- a. An error, and nonconservative assumption, exists in the forced ventilation hot gas layer model, predicting low hot gas layer temperatures.
- b. Radiative heat transfer directly above the flame is not modeled, yielding cooler temperatures directly above the flame than off to the side of the flame.
- c. Two errors in the calculation of view factors overpredict the heat radiated to targets to the side as compared to objects directly above the flame.
- d. Only convective heat transfer, and not the dominant radiative heat transfer for objects directly engulfed in the flame, is modeled. Time to ignition is highly non-physical.
- e. The conduction algorithm is unstable, often resulting in premature termination of the code, especially for cases involving objects in the flame or thermal response of barriers.
- f. The mass burning rate of burning objects is underpredicted due to lack of thermal feedback modeling.
- g. Cable insulation ignition and damage failure threshold criteria are not currently well understood and the results are quite sensitive to the input parameters chosen.

Both small and large fires were postulated in the calculations for N-Reactor. A small fire was assumed to be 2 ft (0.61 m) in diameter and consist of 1 gallon (3.8 l) of oil. A large fire was assumed to be 3 ft (0.91 m) in diameter and consist of 10 gallons (38 l) of oil. Analysis of a data base of transient combustible fuel sources found at nuclear power plants (Ref. 12) indicates that oil sources less than or equal to 1 gallon (3.8 l) were found approximately 70 percent of the time. Oil sources larger than this were found approximately 30 percent of the time. A similar partitioning between small and large quantities in terms of heat content (BTU or KJ) can be made for other credible transient combustible sources such as solvents or trash paper. Again, analysis indicates that a 70/30 partitioning between small and large fuel sources is appropriate (within ± 10 percent). It can also be shown that 10 gallons (38 l) of oil bounds any large solvent or trash paper combustible source in terms of heat content and is, therefore, an appropriate upper bound on transient combustible fuel source size.

A walkdown of the N-Reactor plant was performed to obtain vital information for the COMPBRN calculations. This information included the location of critical equipment and cable trays, separation between redundant trains, types of cable present, and any shielding or fire barriers that may be present. Several "pinch points" were identified where critical cables from redundant trains passed from one room to another, or where cables for important primary/secondary systems and emergency systems were routed through the same area in close proximity. Some of the cabling was enclosed in conduit and much of the exposed cabling in trays was coated with fire retardant coatings. Cable in conduit was assumed incapable of ignition; however, damage was assumed to occur when the surface temperature reached the temperature corresponding to cable failure. Because of its low melting point, the aluminum conduit was neglected for conservatism. Ignition temperatures for coated cables were set high to simulate the demonstrated ability of these coatings to delay ignition and then to limit flame propagation.

Cable insulation and damage thresholds are currently not well known (Ref. 13). For this study, a cable insulation ignition temperature of 773°K (932°F) was assumed along with a damage temperature of 623°K (662°F). For the large fire simulations these thresholds are not as critical to the fire damage time calculations because of the intensity of the flames.

A list of input parameters for the COMPBRN calculations is shown in Table 5.8. These parameters were selected based on past fire analyses at commercial nuclear facilities to represent typical qualified cable insulation. It was assumed that the cabling in the areas of interest at N-Reactor included typical brands of nuclear qualified cable insulation materials, such as Rockbestos Firewall III, Brand Rex, or Okonite. Because of the good flame resistance properties of these cables, no self-ignited (electrically initiated) cable tray fires were postulated. Older cable coated with fire retardant coatings is considered to have similar flame resistance characteristics.

The COMPBRN results are shown in Table 5.9 for the critical areas noted in Section 5.2. Summary discussions of each fire area are provided in subsections 5.6.1 through 5.6.21. A number of scenarios were considered for many of these areas. In most cases, a "zone of influence" was determined for the equipment and fire sizes modeled. In other words, the fire location was varied in the COMPBRN models to determine the maximum distance the fire could be away from the critical equipment and still cause damage. The times to damage increase exponentially as the fire distance increases. For both the small and large fires modeled a best estimate floor site as well as time to damage estimate was used. Using these results, the floor area in which a fire would have to occur to damage critical cables can be estimated. An area ratio can then be calculated by dividing this area by the total floor area of the room, fire area, or building (as appropriate). This reduction factor can then be multiplied by the initiating frequency to estimate the frequency of fires which occur in a critical portion of a given room.

Table 5.8

Modified COMPBRN III Input Parameters

<u>Cable Insulation Parameters</u>	
Density	1715 kg/m ³
Specific Heat	1045 J/kg-K
Thermal Conductivity	0.092 W/m-K
Heat of Combustion	1.85-2.31E-7 J/kg
Combustion Efficiency	0.6-0.8
Critical Temperature	
Piloted Ignition	773°K
Spontaneous Ignition	773°K
Damage	623°K
Surface Controlled Burning Rate	0.0001-0.0075 kg/m ² -S
Burning Rate Radiation Augmentation	1.86E-7 kg/J-m ²
Radiative Fraction	0.3-0.5
Smoke Attenuation Factor	1.4
Reflectivity	0.1-0.3
<u>Oil Parameters</u>	
Density	900 kg/m ³
Specific Heat	2100 J/kg-K
Heat of Combustion	4.67E7 J/kg
Combustion Efficiency	0.9
Surface Controlled Burning Rate	0.06
Radiative Fraction	0.3-0.5
Mass of Oil	3.4-34.0 kg

Table 5.9
Summary of COMPBRN Results

<u>Fire Area</u>	<u>Location</u>	<u>Large Fire</u>		<u>Small Fire</u>	
		<u>Time</u>	<u>Area Ratio</u>	<u>Time</u>	<u>Area Ratio</u>
3A	Room 6 V-3 cabling	6-8 min	0.013	3-4 min	0.008
3A	Room 6 V-4 cabling	11-12 min	0.008	--	--
3A	Room 6 All Systems	4-6 min	0.005	--	--
3B1	Corridor 1 All Systems	10-11 min	0.082	--	--
1D	Outlet Piping	8 min	0.096	4 min	0.051
3B	Room 9	2 min	1.0	--	--
3B	Room 8	5-6 min	1.0	--	--
3B1	Corridor 1 V-4 cabling	12 min	0.070	--	--
3G	Room 30	6 min	0.132	4 min	0.069
3J	Corridor 5	4 min	0.115	4 min	0.052
3B	Room 2	10-12 min	0.190	--	--
3B	Tunnel 47	2-4 min	1.0	3 min	0.130
21	Corridor 19	2-6 min	0.010	--	--
22	Cell Access	6-10 min	0.015	3-4 min	0.003
27B	Bldg. 153 "B"	4 min	0.080	--	--
27A	Bldg. 153 "A"	8 min	0.037	--	--
31A	Bldg. 182 Terminal Box	--	--	--	--
31A	Bldg. 182 HPI Pumps	--	--	--	--
31B	Bldg. 182 LPRW Pumps	--	--	--	--
32A	Bldg. 184	4-6 min	0.001	--	--
32A	8 Ft Tray 16 Ft Tray	4 min --	0.021 --	4 min --	0.009 --

It should be noted that a small fire does not yield damage in many of the areas. Prior experience with COMPBRN shows that a small fire must be very close to its target to yield damage. Large fires, however, can and do yield damage in most of the areas. The major exception is in small closed rooms (like the battery room) in which a hot gas layer rapidly develops. In such cases, the hot gas layer effects become quite significant. Thus, for some of the COMPBRN runs, room parameters were used in order to simulate a model of the hot gas layer. For these cases, damage occurs sooner due to the increased thermal input from the hot gases.

It has been found in past experience with COMPBRN and in some of the simulations for N-Reactor that the COMPBRN results can be quite sensitive to fires located adjacent to walls which are in close proximity to the target cable trays. Using the typical model of the wall as one section results in unrealistic radiative heat fluxes from the wall to the cable trays of interest. For these cases, the wall was divided into several sections to more realistically calculate the wall thermal response.

The following subsections discuss the approach and results for the scenarios analyzed in each of the critical fire areas.

5.6.1 Fire Area 3A: Room 6, V-3 Valves

5.6.1.1 Results

The large fire simulation resulted in damage to cable tray TR-23 in 6 to 8 minutes (time differences occurred due to COMPBRN sensitivity to geometric configuration) for a small area around the cable trays in the northwest corner of the room. Using 600 m² as the area of the cable spreading room, yields an area ratio of 0.013 for a large fire. A small fire can do damage to TR-23 in 3 to 4 minutes with an area ratio of 0.008.

5.6.1.2 Discussion

The angled set of cable trays was "linearized" and modeled as a straight section of trays due to COMPBRN limitations, and the walls at the corner were modeled along with the ceiling to simulate a hot gas layer. No other trays were modeled and the runs of trays along the north and west walls were ignored, as it was not certain where cables were routed once they entered these trays from TR-23. At the outer limits of the "zone of influence," the above results occur with no actual ignition of cables. Assuming that damage can occur within these times regardless of the Flamastic coating, there was no need to take credit for the cable coating for this simulation.

5.6.2 Fire Area 3A: Room 6, V-4 Valves

5.6.2.1 Results

For the large fire and taking no credit for Flamastic, damage and ignition occurred. Taking credit for Flamastic to prevent ignition and flame propagation yielded damage in 11 to 12 minutes with an area ratio of 0.008. For a small fire, assuming Flamastic and no ignition, damage alone to the target tray will not occur.

5.6.2.2 Discussion

Cables for the V-4 valves are routed through tray TR-31 which is the lower and most southerly tray entering the cable spreading room in the southwest corner. Due to the limited space, some of the other lower adjacent trays located in this area and which make a bend at the room corner were also modeled. Again, the walls and ceiling were modeled using parameters to simulate the hot gas layer.

Due to the general use of Flamastic in this room and the rest of the plant, it was deemed appropriate to take credit for it, a comment which applies to the remainder of this report. It was not possible to actually model Flamastic-like cable coatings with the current or modified versions of COMBURN. Instead, the ignition temperatures were set high enough to prevent actual cable ignition, a decision supported by actual experimental results which show that these coatings are quite effective in delaying ignition or limiting flame propagation once ignition does occur. This approach limits the analysis to the fire sizes and locations which result in cable insulation damage only.

5.6.3 Fire Area 3A: Room 6, "All Systems"

5.6.3.1 Results

Taking credit for the Flamastic coating and assuming no ignition in this congested area, a large fire will result in damage to the target cable tray(s) in 4 to 6 minutes. A small fire cannot damage these cables.

Assuming that the critical cables are located in both of the lower sets of trays in the two trains of cable trays entering the room from Corridor 1, the large fire area ratio (for Room 6) would be 0.001. If it is assumed conservatively that the critical cables could be located in either of the lower trays, then the area ratio becomes 0.005. The latter case was chosen for this analysis, since the actual locations of the cables was not traced and is therefore unknown.

5.6.3.2 Discussion

Both sets of trays, two layers deep (due to limitations on the number of modules that can be modeled), were modeled in addition to four layers of lower adjacent trays in the run of trays along the south wall. It was

necessary to "partition" the south wall and to move the fire to numerous locations due to the congestion in this area of the room. As the fire location was constrained to the area immediately below and adjacent to the trays, ignoring the protection provided by the Flamastic would result in significant ignition and flame propagation and a much larger, over-conservative area ratio. Therefore, it was judged of particular importance to take credit for the cable coating in this area.

5.6.4 Fire Area 3B1: Corridor 1, "All Systems"

5.6.4.1 Results

The configuration of the cable trays is quite similar to the previous discussion. Again, no small fire model will damage these cables. The large fire, assuming Flamastic protection, yields damage to both lower cable trays in 12 to 13 minutes with an area ratio of 0.024 for the corridor. As above, assuming that critical cables could be in either of the 2 lower trays, damage will occur in 10 to 11 minutes with an area ratio of 0.082.

5.6.4.2 Discussion

This long, narrow, closed corridor was modeled in its entirety, using hot gas layer parameters and involved partitioning the "front" (north) and "back" (south) walls due to the close proximity of the fire to both. As was the case on the other side of the wall in Room 6 (discussed above), the first two layers of trays in both rows of cable tray runs were modeled and the fire was of course constrained in its movement axially along the corridor in determining the zone of influence.

5.6.5 Fire Area 1D: Outlet Piping Space, V-4 Valves

5.6.5.1 Results

A large fire could damage the cable in conduit for the V-4 valves in this area in 8 minutes over an area ratio (based on the floor area of this room) of 0.096. A small fire will damage the cable in conduit in 4 minutes with an area ratio of 0.051.

5.6.5.2 Discussion

Information from plant personnel was used to estimate the location of this cable run. It was assumed that the common conduit run was approximately 2.75 meters above the floor level and about 75 percent of the length of the room. The fire location was varied normal to the axis of the conduit run to determine the zone of influence either side of the conduit for its entire length.

5.6.6 Fire Area 3B: Room 9, V-4 Valves

5.6.6.1 Results

A large fire results in damage to the conduit-enclosed cables in 2 minutes from anywhere in the room, due to the pronounced hot gas layer effects in this small room (area ratio of 1.0). A small fire cannot damage these conduit runs.

5.6.6.2 Discussion

The four parallel, nonadjacent conduit runs were modeled across the north end of this room. Both the north wall and the ceiling were modeled, along with the hot gas layer. Due to the small closed nature of this room, the hot gas layer is a dominant factor in the results.

5.6.7 Fire Area 3B: Room 8, V-4 Valves

5.6.7.1 Results

A large fire will damage the cable tray runs from anywhere in the room (area ratio of 1.0) in 5 to 6 minutes. A small fire cannot result in damage.

5.6.7.2 Discussion

As with Room 9, this small room required hot gas layer modeling. Two sections of the cable tray run were modeled to note any effects from fires located in different parts of the room. Due to the small closed nature of this room, the hot gas layer is a dominant factor in the results.

5.6.8 Fire Area 3B1: Corridor 1, V-4 Valves

5.6.8.1 Results

Large fire damage occurs to the lower corner of this cable tray run in 12 minutes from anywhere across the width of the corridor and axially along the corridor to an extent which yields an area ratio on Corridor 1 of 0.07. As no ignition occurs, the separate case involving Flamastic was not needed. No small fire can damage this tray run.

5.6.8.2 Discussion

The room setup for this scenario was similar to the Corridor 1 configuration discussed in subsection 5.6.4 above. The fire is modeled quite low as the actual floor is below the grated walkway another 4 ft. Both the front and back wall sections were partitioned, along with models of the ceiling and hot gas layer. The corridor was "linearized" to yield an effective ceiling area and overall fire area floor area. Because of the difference in corridor width in the two sections, the corridor was

made into an equivalent corridor 10 ft wide for this scenario. Due to the length of the room and remoteness of the ceiling midpoint, this approach was considered reasonable. Once setup, the above model yielded reasonable results though the modeling was not as straightforward as in other areas. The actual fire location would be somewhat restricted due to the walkway. However, to be conservative, the area ratio was determined assuming a fire can occur anywhere across the corridor.

5.6.9 Fire Area 3G: Room 30, V-3 Valves

5.6.9.1 Results

When the delaying effects of Flamastic are considered (i.e., no ignition was allowed to occur for the typical time frames involved in the COMPBRN simulations), a large fire yields damage in 6 minutes over an area ratio for this room of 0.132. The small fire will damage the cables in 4 minutes with an area ratio of 0.069. This analysis was completed prior to changing the elevator oil to cosmopolubricant HF-130.

5.6.9.2 Discussion

This is a "hot" area, both in terms of room conditions and in the amount of available combustibles. All of the trays were modeled in the actual matrix involving two parallel rows of four trays stacked vertically each, with the target TR-23 being the second from the bottom in the "back" row next to the wall. The south wall was partitioned and the ceiling and hot gas layer effects were modeled. Due to the somewhat "hidden" location of the target tray relative to the flames, it could be judged a marginal assumption to ignore the potential ignition of the cable trays below and adjacent to this tray. Actual ignition, if sustained and of sufficient heat output, could increase the floor area that the large fire could cause damage from. However, it was judged that even should ignition occur, there would likely not be a significant increase in additional heat to the target tray from the other trays due to the likely low burning rate, as seen from experimental results.

5.6.10 Fire Area 3J: Corridor 5, V-3 Valves

5.6.10.1 Results

Taking credit for the Flamastic coating, a large fire in this area will damage the target cables in 4 minutes with an area ratio figured on Corridor 5 of 0.115. A small fire will cause damage in 4 minutes with an effective area ratio of 0.052.

5.6.10.2 Discussion

This was a relatively simple run to model, due to the open corridor, with no need to model walls, ceiling, or hot gas layer parameters. TR-23 is more exposed to the fire here than in Room 30, not being backed up against the wall or shielded as much by other trays. Also, there were significant flammables in this area.

5.6.11 Fire Area 3B: Room 2, V-3 Valves

5.6.11.1 Results

The large fire can damage the cables in TR-23 in 10 to 12 minutes within 1.5 meters of the wall, for an area ratio (for Room 2) of 0.190. The small fire did not result in damage temperatures.

5.6.11.2 Discussion

This model was kept simple due to the high ceiling, lack of combustibles, and openness of this room. It was not necessary to model the ceiling or hot gas layer, though the south wall was segmented due to the need to locate the fire right adjacent to it.

5.6.12 Fire Area 3B: Electrical Tunnel 47, V-3 Valves

5.6.12.1 Results

A large fire will damage the TR-23 cables in 2 to 4 minutes, depending on location along the corridor, regardless of whether credit is taken for Flamastic. The area ratio on the corridor itself is 1.0.

A small fire can do damage to the trays midway down the length of the tunnel where the trays come closest to floor level. Damage occurs in 3 minutes for an area ratio of about 0.13. More extensive damage over a larger area is possible with a small fire simulation when the Flamastic coating is ignored. However, the extent of ignition that occurs is highly unlikely for the actual installation, given the experimental results available for cable coatings.

5.6.12.2 Discussion

This area was extensively modeled due to the high, narrow nature of the tunnel and the varying heights of the cable trays from the floor. The hot gas layer and ceiling were not modeled as this area is open to Room 1 and Corridor 5 via the cable chase. Three locations were modeled due to the varying heights of the lower tray from the floor. The walls were modeled with the wall behind the cable trays segmented. The target tray TR-23 was the eighth tray up in the stack of trays. Due to the height of the tray of interest, it takes a high flame from a large fire to damage it and not also cause ignition to the coated cables in very short time frames. Except at mid-tunnel where the trays are closest to the floor, the small fire was not able to get the eighth tray up hot enough to cause damage unless actual ignition was allowed to occur and extensively propagate up the lower levels of trays from the start of the simulation. This was not considered a good physical representation; therefore, only the model using Flamastic coatings was used to determine results.

5.6.13 Fire Area 21: Corridor 19, "All systems"

5.6.13.1 Results

This scenario was split into two segments; the section of the trays in the turbine drive bay basement area, and the trays in Corridor 19 itself which are isolated from the former by a closed fire door. In neither case can a small fire result in damage to the lower trays.

In Corridor 19, the large fire can cause damage from anywhere in the corridor (area ratio on the corridor of 1.0) in 2 to 3 minutes. The effective area for fire damage was reduced by taking partial credit for the sprinkler system.

In the basement area, a large fire can cause damage over a gross floor area of over 63 m², which again was reduced by taking partial credit for sprinkler suppression effectiveness. The overall area ratio for Building 109 with these results is 0.010.

5.6.13.2 Discussion

The first two layers of trays in all three rows of trays in both areas were modeled. Due to the open nature of both areas, the hot gas layer was not modeled. The walls in Corridor 19 were modeled, and were segmented vertically due to the nearness of the fire and the height of the walls. This segmentation has a significant affect and was considered quite important towards realization of physically meaningful results. It was assumed that the critical cables could be located in any one of the lower trays. At the outer bounds of the fire locations, typically two and sometimes more of the cable trays saw temperatures which resulted in cable damage. Therefore, this conservative approach was considered reasonable. The actual spacing of the trays was determined from the drawings, as it varied along the cable tray length from the basement (drawing sections AA-AA and B-B) to the corridor (section X-X).

5.6.14 Fire Area 22: Cell Access Corridor, PT-260, PT-261

5.6.14.1 Results

The large fire could damage these transmitters in 6 to 10 minutes depending on the direction the fire is from the actual location of the transmitters. The resultant area ratio on the corridor itself is 0.015. The small fire causes damage in 3 to 4 minutes with an area ratio of 0.003.

5.6.14.2 Discussion

The actual location of the transmitters was based on plant personnel recall, as this area was not accessible for inspection. To be conservative, the transmitters were placed halfway down the length of the corridor at the ceiling midpoint. Since this corridor is usually closed, the hot gas layer was modeled. The wall the transmitters are mounted on

was modeled in segments. It was assumed that the transmitters are mounted about 5 ft (1.5 meters) off the floor. Note that this scenario does not involve Flamastic considerations.

5.6.15 Fire Area 27B: Building 153, Instrument Air Compressor Conduit Run

5.6.15.1 Results

A large fire damages this conduit run in 4 minutes with an area ratio of 0.080. A small fire will not damage this run of cable in conduit.

5.6.15.2 Discussion

This scenario was relatively simple to model and included provisions for the hot gas layer, the ceiling, and no walls. The results included the area ratio for a second run of conduit which penetrates from the "A" side.

5.6.16 Fire Area 27A: Building 153, Instrument Air Compressor Conduit Run

5.6.16.1 Results

No damage is caused by a small fire. The large fire will damage the cable in 8 minutes with an area ratio of 0.037.

5.6.16.2 Discussion

As discussed in the previous subsection, this scenario includes hot gas layer parameters and involves a simple cable conduit run from the switchgear to the wall penetration to the "B" side.

5.6.17 Fire Area 31A: Building 182, Master Terminal Box

5.6.17.1 Results

The large and small fires typically modeled yielded no damage to any of the cable in conduit, trays, or the terminal box which were targeted for this scenario. It required a fire 1.67 times larger in surface area than the large fire to damage the target cables.

5.6.17.2 Discussion

The actual Master Terminal Box was not located correctly during the field inspection. It's correct location was noted on the drawings and later verified in the field. The drawings provided enough detail to set up a fairly elaborate model of the terminal box and the cable entering (from a cable tray) and leaving (in conduit). The adjacent wall, hot gas layer parameters, and ceiling were modeled, providing considerable conservatism.

5.6.18 Fire Area 31A: Building 182, HPI Pumps

5.6.18.1 Results

No damage to any three HPI pumps occurs due to either large or small fires. It would take a fire over seven times larger than the large fire simulation typically modeled to yield damage.

5.6.18.2 Discussion

The actual drawing dimensions between the closest three HPI pump motor leads (in vertical conduit) were modeled. The possible shielding effects of the motors were ignored as the motors themselves could be the actual target for fire damage. In this case, no ceiling, walls, or hot gas layer effects were modeled.

5.6.19 Fire Area 31B: Building 182, LPRW Pumps

5.6.19.1 Results

No damage is caused due to either large or small fires, to all three pumps. It would take a fire with a pool area over 2.76 times as large as the large fire to damage all three pump motors or motor leads.

5.6.19.2 Discussion

Cable in conduit from the motor junction box down through the floor was modeled. The scenario was similar to that modeled for the HPI pumps, with no motor shielding effects recognized, and no ceiling, walls, or hot gas layer modeled.

5.6.20 Fire Area 32A: Building 184, Instrument Air Compressors

5.6.20.1 Results

A large fire will damage the conduit runs from the compressors in 4 to 6 minutes with an area ratio for the net area of the basement (discounting the boiler floor area itself) of 0.001. A small fire will not damage both of the compressors at the same time.

5.6.20.2 Discussion

This scenario was similar to the previous two, with two vertical runs of conduit, no ceiling, walls, or hot gas layer, and the fire varied in location between the two compressors.

5.6.21 Fire Area 32A: Building 184, Instrument Air Compressor Cable Trays

5.6.21.1 Results

This scenario was divided into two cases, involving cable tray runs at different heights. For the 8-ft-high tray, the large fire will damage the cables in 4 minutes with an area ratio of 0.021. The small fire can cause damage in 4 minutes over an effective area ratio of 0.009.

No damage occurs to the cables in the tray located 16 ft above the floor, using either large or small fire simulations. A separate analysis determined that a pool 1.67 times larger than the large fire would be needed to damage this run of cables.

5.6.21.2 Discussion

No wall, ceiling, or hot gas layer parameters were modeled. The actual tray run lengths were used in the determination of area ratio, and the gross floor area of the basement of 184 was reduced by the boiler area to yield a net area for calculations. Note that use of Flamastic was not noted in this area but was not of significance anyway.

5.7 Barrier Failure Analysis

In the unscreened cut sets where a potential for barrier failure had been identified, barrier failure probability was estimated using barrier failure rates developed as part of the RMIEP program (Ref. 14).

Barriers were grouped into three types: (1) fire doors, security doors, water-tight doors, and fire curtains, (2) fire dampers and ventilation dampers; and (3) penetration seals and fire walls. The data base contains 628 records from when construction began on any given plant to the end of June 1985. The number of barriers of each type at a plant is required to estimate the rate at which a specific component fails. The number is not known precisely for each plant, but a nominal figure that has been estimated for each barrier type is given in Table 5.10.

Table 5.10

Approximate Number of Barriers at a Plant

<u>Type</u>	<u>Nominal</u>
1	150
2	200
3	3000

The statistical uncertainty of each estimate, reflecting sampling variation and plant-to-plant variation, is represented by 90 percent confidence bounds and was developed using a binomial failure rate model. These estimates and confidence bounds are given in Table 5.11 where units of both estimates and bounds are failures/year.

Table 5.11
Estimates of Single Barrier Failure Rate

<u>Barrier Type</u>	<u>Barrier/ Unit</u>	<u>Estimate</u>	<u>5% Confidence Bound</u>	<u>95% Confidence Bound</u>
1	150	7.4E-3	0.0	2.4E-1
2	200	2.7E-3	0.0	2.2E-1
3	3000	1.2E-3	0.0	3.7E-2

During the confirmatory plant visit scenarios which required barrier failure had those barriers inspected. No plant specific vulnerabilities were noted as a result of this inspection which would require modification of generic barrier failure rates. After multiplying barrier failure rates by the number of penetrations at each appropriate fire area adjacency and utilizing the probabilities developed in screening step 5, all remaining barrier failure scenarios did not survive the 10^{-8} per year frequency screening criteria.

5.8 Recovery Analysis

For those remaining cut sets which survived the screening process and where the COMPBRN code predicted fire damage would occur, recovery of random failures and credit for extinguishment of the fire before the COMPBRN predicted time to fire damage was applied. A detailed discussion of the quantification procedure for the probability of extinguishment before fire damage occurs is contained in Appendix A.

Recovery of random failures (non-fire related) was treated in a similar fashion as in the internal events analysis. All operator recovery actions that were used in the internal events analysis were inspected for use where appropriate on the remaining cut sets. If a sequence was long-term (greater than 24 hr), two recovery actions were allowed. In short-term (less than 24 hr) sequences only one recovery action was allowed.

This recovery action was chosen if the possibility of multiple recovery actions was present and on a hierarchy based on recovery probabilities established by the internal events analysts.

In the areas where firefighting activity takes place, no credit was given for local recovery actions until after the fire was extinguished. In non-affected areas, local recovery was allowed for valve manipulation or pump operation when damage to power cabling of an applicable component had not occurred.

5.9 Uncertainty Analysis

Distributions on fire frequency, fire suppression probability, fire code calculations, random failure probability, area and severity ratios, and operator recovery actions, generated uncertainties on fire-induced core damage frequencies.

The uncertainty of these values was propagated through the accident sequence models using two computer codes. A Latin Hypercube Sampling (LHS) algorithm was used to generate the samples for all of the parameter values (Ref. 15). The Top Event Matrix Analysis Code (TEMAC) was used to quantify the uncertainty of the accident sequence equation using the parameter value samples generated by the LHS code (Ref.16).

LHS is a constrained Monte Carlo technique which forces all parts of the distribution to be sampled. The LHS code is also flexible in that it can sample a variety of random variable distributions. Furthermore, parameter distributions for similar events were correlated. For example, if two similar components (e.g., MOV XX-FTO and MOV YY-FTO) are modeled from the same probability distribution, then the sampling of these two distributions is perfectly correlated, meaning the same value is used for both events in a given sample member. For basic events which are modeled with very similar but slightly different distributions (e.g., MOV XX fails to remain closed for 100 hr and MOV YY fails to remain closed for 200 hr), the LHS code permits an induced correlation between the samples. However, LHS does not allow the correlation coefficient for this case to be equal to 1.0. LHS does permit sampling with a coefficient of 0.99 in these cases.

TEMAC uses the LHS parameter samples and the accident sequence equations (cut sets) as input to quantify the core damage estimates. TEMAC generates a sample of the accident sequence frequency, a point estimate of the frequency, and various importance measures and ranking for the base events.

Uncertainty on fire initiating event frequency was developed when the generic fire frequencies were updated using N-Reactor specific data. This process which was briefly discussed in Section 5.3 is covered in more detail in Modeling Time to Recovery and Initiating Event Frequency for Loss of Off-Site Power Incidents at Nuclear Power Plants (Ref. 5).

Uncertainty on fire non-suppression probabilities ($Q(\tau_G)$) was addressed by modification of COMPBRN predicted time to damage. The COMPBRN predicted time to damage and its associated non-suppression curve probability were taken to be a best estimate of a maximum entropy distributed variable. Fifteen minutes were added and subtracted from the COMPBRN predicted time to allow for uncertainty in its result and the uncertainty in the probability of non-suppression distribution. These probabilities were taken as a minimum and maximum of the maximum entropy distribution respectively.

Uncertainty associated with the fire size estimate factor (f_s) was developed utilizing information associated with an I&E inspector report (Ref. 12) on a survey of different types of combustibles and their amounts found in nuclear power plants. Two fire sizes, a large and small fire were modeled as described in Section 5.6. These fire sizes (BTU content) were compared to the distributions on possible fire sizes developed for the different combustibles from the I&E data. The best estimate for percentage of fires that were either large or small was taken from an average of the different types of combustibles for an equivalent BTU level fire modeled by COMPBRN. This probability was assumed to be the best estimate (mean) value of a maximum entropy distribution. For a complete description of the maximum entropy distribution refer to Reference 16. Maximum and minimum probabilities for this distribution were assumed to be based on one individual type of combustible with either the maximum and minimum percentage corresponding to applicable fire size (BTU rating).

Random failure events and operator recovery actions were treated identically as in the internal events analysis. Uncertainties and types of distributions were not modified for the fire analysis.

All other factors and their associated uncertainties are not common to all fire sequences and will be addressed individually in the appropriate subsections of Section 5.10.

5.10 Description of Unscreened Fire-Induced Core Damage Scenarios and Their Associated Fire Areas

5.10.1 Introduction

This section will describe the fire scenarios and their associated fire areas which are listed in Table 5.1. All other fire areas and all adjacent fire area combinations dropped below 10^{-8} per year after either operator recovery of non-fire-related failures, COMPBRN code calculations, or barrier failure probabilities were applied.

The base case analysis assumes modifications to the plant that effect fire scenarios that will be described in corridor 1, corridor 19, and the control room.

The modification to corridor 19 and 1 consists of installation of an automatic sprinkler system that has full area wide coverage to suppress any transient combustible fire originating below the critical cable trays. In the case of the control room, it was assumed that the alternate shutdown panel would be fully operational and procedures would be in place for its use in the event of a control room fire.

The overall effect of these assumptions was an approximate order of magnitude reduction in total fire-induced core damage frequency.

5.10.2 Cell Access Corridor

The cell access corridor is in the basement of Building 109 immediately adjacent to Corridor 19. Fire-related damage in this corridor was to transmitters PT-260 and PT-261. Failure of these transmitters led to failure of the PCS/SCS (G). The emergency core cooling system failed randomly (failure not related to the fire itself) to cool the entire core (A_2). The GSCS (C) was not affected by this scenario and thus a fire-induced transient ($GA_2\bar{C}$) sequence arose.

It was found that both a large and small fire could damage the transmitters; thus both severity factors (f_{S1} for a large fire and f_{S2} for a small fire) are incorporated in the core damage equation. Uncertainty for both of these factors was developed as described in Section 5.9.

For both sizes of fire an area ratio factor (f_{A1} for a large fire and f_{A2} for a small fire) was determined based on the COMPBRN results of Section 5.6. This area ratio factor is defined as that percentage of area within the fire area where fire damage to PT-260 and 261 could occur for either a large or small fire. Another area ratio factor f_{22} subdivides Building 109 fire frequency to that percentage that will occur in the cell access corridor. In all three cases these ratios were assumed to be best estimates of a maximum entropy distribution. Upper and lower bounds on these distributions were taken to be factors of five higher and lower than the best estimate probability, respectively.

The term $Q_1(\tau_G)$ which assesses the probability of manual non-extinguishment of the fire before damage occurs to PT-260 and PT-261 was developed as described in Section 5.9 using the information developed on fire brigade response times from Appendix A.

The frequency of fire in Building 109 (λ_{109}) was developed as described in Section 5.3 assuming that Building 109 could most closely be approximated by a commercial reactor turbine building. Thus, in the first Bayesian updating stage commercial LWR turbine building fire data were utilized. For a more detailed explanation of the Bayesian updating procedure used in this analysis refer to the discussion in Section 5.3.

The factor A_2 that appears in the core damage equation was solved using the internal events fault tree for failure to cool one-sixteenth of the core due to random check valve failure. None of the basic events used in the solution of A_2 had any fire-related failure mechanism. Once this probability was quantified, the cut sets used in the solution of A_2 were collapsed to a single variable which was lognormally distributed with an error factor of 10.

Thus, the core damage equation for the cell access corridor is as follows:

$$\phi_{cm} = \lambda_{109} f_{22} Q_1(\tau_G) A_2 [f_{A1} f_{S1} + f_{A2} f_{S2}]$$

where,

ϕ_{cm} = fire-induced core damage frequency for the cell access corridor

λ_{109} = frequency of Building 109 fires

f_{22} = area ratio of the cell access corridor to that of Building 109

$Q_1(\tau_G)$ = probability of the fire brigade not suppressing fire before damage occurs

A_2 = random failure of one check valve and thus failure to cool one-sixteenth of the core

f_{A1} = ratio of area within the cell access corridor for a large fire

f_{S1} = severity ratio for a large fire

f_{A2} = ratio of area within the cell access corridor for a small fire

f_{S2} = severity ratio for a small fire

Table 5.12 gives the values of each of these factors as well as their associated distribution and upper and lower bounds. For all lognormal and gamma distributed variables in Table 5.12 and the following tables of Section 5.10, the lower and upper bound are the 5th and 95th percentiles of the distribution while the best estimate is the mean value. All final results for the respective core damage equations were presented in Table 5.1.

Table 5.12

Cell Access Corridor Fire Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{109}	Gamma	0.022	0.043	0.11
f_{22}	Maximum Entropy	7.0E-3	0.036	0.18
$Q_1(\tau_G)$	Maximum Entropy	0.40	0.99	1.0
A_2	Lognormal		2.27E-3	(EF = 10)
f_{A1}	Maximum Entropy	3.0E-3	0.015	0.075
f_{A2}	Maximum Entropy	6.0E-4	3.0E-3	0.015
f_{S1}	Maximum Entropy	0.19	0.30	0.67
f_{S2}	Maximum Entropy	0.33	0.70	0.81

5.10.3 Building 153 Main Floor "B" Switchgear

Building 153 contains 4.1kV switchgear for both the "A" and "B" buses. The 13.8kV switchgear is also located on the "B" bus side of the ground floor.

This scenario is similar to that of Section 5.10.2 in that fire-induced damage fails the PCS/SCS (G), the ECCS (A_2) randomly fails, and the GSCS succeeds (G). Thus, once again transient sequence ($GA_2\bar{C}$) occurs. For this plant area a fire damages cabling enclosed in conduit for both the instrument air and service air compressors. Failure of two out of three of the two instrument air compressors and one service air compressor will fail the PCS/SCS.

It was found that only a large fire can damage these cables, therefore, the severity ratio factor is for large fires only (f_{S1}).

The term $Q_2(\tau_G)$ which assesses the probability of manual extinguishment before fire damage occurs is quantified in an identical fashion as the corresponding term in Section 5.10.2. The area ratio term (f_{A3}) is that percentage of the main floor area (B side) of Building 153 where a large fire can damage the compressor's cabling. Uncertainties are developed in

similar fashion to those area ratios in Section 5.10.2 where upper and lower bounds on the maximum entropy distribution were taken to be a factor of five difference from the best estimate value.

The frequency of Building 153 "B" side ground floor fires (λ_{SGR}) was developed assuming that this area most closely resembled a commercial LWR 4160V switchgear room area. Thus, in the first Bayesian updating stage switchgear room fire data were utilized.

Thus, the core damage equation for this area is as follows:

$$\phi_{cm} = \lambda_{SGR} f_{A3} f_{S1} Q_2(\tau_G) A_2$$

where,

ϕ_{cm} = fire-induced core damage frequency for Building 153 main floor "B" side

f_{S1}, A_2 are identical to those terms described in Section 5.10.2

f_{A3} = ratio of area within the Building 153 ground floor "B" side for a large fire

λ_{SGR} = frequency of Building 153 main floor "B" side fires

$Q_2(\tau_G)$ = probability of the fire brigade not suppressing the fire before damage occurs

Table 5.13 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.13

Bldg. 153 Main Floor ("B" Side) Fire Scenario
Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{SGR}	Gamma	6.2E-6	2.8E-3	5.9E-3
f_{A3}	Maximum Entropy	0.016	0.08	0.40
f_{S1}	Maximum Entropy	0.19	0.30	0.67
A_2	Lognormal		2.27E-3	(E.F. = 10)
$Q_2(\tau_G)$	Maximum Entropy	0.38	0.99	1.0

5.10.4 Building 153 Main Floor "A" Switchgear

The "A" switchgear portion of Building 153 is immediately adjacent to the "B" side of the building. These two rooms were divided into separate fire areas because of a dividing wall and fire doors.

The scenario for this area is identical to that of the previous section. A transient sequence (GA_2C) results from the fire-related failure of the instrument air system. The only modification to the core damage equation is in the area ratio factor (f_{A4}).

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{SGR} f_{A4} f_{S1} Q_2(\tau_G) A_2$$

where,

ϕ_{CM} = fire-induced core damage frequency for Building 153 main floor 'A' side

λ_{SGR} , $Q_2(\tau_G)$, A_2 , f_{S1} are identical to those terms described in Section 5.10.3

f_{A4} = ratio of area within Building 153 ground floor 'A' side for a large fire

Table 5.14 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.14
Building 153 Main Floor ("A" Side) Fire Scenario
Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{SGR}	Gamma	6.2E-6	2.8E-3	5.9E-3
f_{A4}	Maximum Entropy	7.4E-3	0.037	0.19
f_{S1}	Maximum Entropy	0.19	0.30	0.67
A_2	Lognormal		2.27E-3	(E.F. = 10)
$Q_2(\tau_G)$	Maximum Entropy	0.38	0.99	1.0

5.10.5 Building 184 Instrument Air Compressor Cable Runs

Building 184 contains all instrument air and service air compressors. Cable trays run from the area on the ground floor where these compressors are located and exit the building on the south side.

As was the case for Sections 5.10.3 and 5.10.4 fire damage to the instrument air system cable runs fails the PCS/SCS and leads to a transient (GA_2C) sequence.

For the cable runs in Building 184 both a large and a small fire are capable of causing damage. Therefore, the core damage equation contains two fire severity factors (f_{S1} and f_{S2}) and floor area ratios associated with both fire sizes. As was the case for the area ratios previously determined the COMPBRN code results aided in their determination. The length of the cable run was from where the cable trays went from being approximately 16 ft above the floor to only 8 ft high and then to the south wall of the building. No fire damage was predicted for the cable trays at 16 ft. The width of influence for the 8-ft-high cable tray was taken directly from the COMPBRN results. These floor areas were then ratioed to the total floor area of Building 184 excluding the annex.

The area ratio factor (f_{32A}) then ratioed the floor area of Building 184 minus the annex to that of total Building 184 floor area.

Building 184 most closely resembled an auxiliary building of commercial reactor plants. Therefore, the term (λ_{184}) which represents the frequency of fire anywhere in Building 184 utilized generic auxiliary building fire data in the first stage of the Bayesian updating procedure.

The term $Q_3(\tau_G)$ which assesses the probability of manual extinguishment before fire damage occurs was quantified in an identical fashion as the corresponding terms in the previous sections.

Thus, the core damage equation for this scenario is as follows:

$$\phi_{CM} = \lambda_{184} Q_3(\tau_G) A_2 f_{32A} [f_{S1} f_{A5} + f_{S2} f_{A6}]$$

where,

ϕ_{CM} = fire-induced core damage frequency for Building 184 instrument air and service air compressor cable runs

$Q_3(\tau_G)$, f_{S1} , f_{S2} , A_2 are identical to those terms previously described in Section 5.10.2

f_{A5} = area ratio for a large fire within fire area 32A

f_{A6} = area ratio for a small fire within fire area 32A

f_{32A} = area ratio of fire area 32A to that of the total floor area of Building 184

λ_{184} = frequency of Building 184 fires

Table 5.15 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

5.10.6 Building 184 Instrument Air Compressor Area

This scenario is identical to that of the previous section except that the fire damage is to the instrument air and service air compressors themselves. Once again, this damage causes a transient ($GA_2\bar{C}$) sequence.

Due to the separation between adjacent compressors (only two of which have to fail due to the fire) only a large fire will cause damage. Therefore, the severity ratio (f_{S1}) and corresponding area ratio (f_{A7}) factors are incorporated as part of the core damage equation. The COMPBRN code results yielded that percentage of fire area 32A (total floor area of Building 184 minus the annex) where the fire could be located and still damage two adjacent compressors.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{184} f_{S1} f_{A7} A_2 Q_3(\tau_G) f_{32A}$$

where,

ϕ_{CM} = fire-induced core damage frequency for Building 184 instrument air compressor area

λ_{184} , f_{S1} , A_2 , $Q_3(\tau_G)$, f_{32A} are identical to those terms described in Section 5.10.5

f_{A7} = area ratio for a large fire within fire area 32A

Table 5.16 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

5.10.7 Building 182 Control Room

Building 182 contains many safety-related frontline and support system pumps. Its control room is located on the ground floor of the building.

One fire scenario in the control room remained after screening and was based on the transient sequence ($GA_2\bar{C}$).

This sequence resulted from fire damage to a control cabinet for many of the support pumps for the PCS/SCS. Credit was given for quick extinguishment of the fire since the control room is almost always manned and also contains an automatic halon system. An analysis of the halon system indicated that testing had been done on a similar system in another plant location and that greater than 6 percent halon concentration was maintained. Also, calculation sheet records were

Table 5.15

Building 184 Instrument Air Compressor Cable Runs Fire Scenario
Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{184}	Gamma	0.03	0.065	0.14
f_{S1}	Maximum Entropy	0.19	0.30	0.67
f_{S2}	Maximum Entropy	0.33	0.70	0.81
$Q_3(r_G)$	Maximum Entropy	0.64	0.99	1.0
f_{32A}	Maximum Entropy	0.14	0.71	1.0
f_{A5}	Maximum Entropy	4.2E-3	0.021	0.11
f_{A6}	Maximum Entropy	1.8E-3	9.0E-3	0.045
A_2	Lognormal		2.27E-3	(E.F. = 10)

Table 5.16

Building 184 Instrument Air Compressor Area Fire
Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{184}	Gamma	0.03	0.065	0.14
f_{S1}	Maximum Entropy	0.19	0.30	0.67
f_{A7}	Maximum Entropy	2.0E-4	1.0E-3	5.0E-3
$Q_3(r_G)$	Maximum Entropy	0.64	0.99	1.0
f_{32A}	Maximum Entropy	0.14	0.71	1.0
A_2	Lognormal		2.27E-3	(E.F. = 10)

reviewed to insure that a proper amount of halon was present to obtain 6 percent concentration level in this area. Since halon testing experience indicates that concentrations of approximately 6 percent should be adequate to extinguish a fire and thorough area coverage by smoke detectors was noted during the plant walkdown, it was assumed that there was no reason to make a modification to halon system reliability estimates. The term (f_R) gives credit for either manual or automatic extinguishment of the fire before damage is sustained to the PCS/SCS support systems control cabinet. Based on the analysts judgement, only one in 100 fires would not be extinguished in time. Halon system reliability was judged to be approximately 95%. In addition, since the control room is continually manned, it was judged that even if the fire was not automatically suppressed that operators would detect and suppress most fires. Therefore, only one in 100 fires were assumed not to be suppressed before smoke forced abandonment of the control room.

The cabinet area ratio factor (f_{A8}) ratioed the area of the PCS/SCS support system control cabinet to that of total cabinet and switchgear area in the control room. Fire data indicate that control room fires have only occurred in electrical cabinets, therefore, it is assumed that a fire occurring in any other portion of the control room floor area has a negligible frequency in comparison to those occurring in electrical cabinets.

As in the previous sections the ECCS (A_2) fails randomly thus leading to a partial core cooling failure.

Commercial control room fire data was used in the first stage of the Bayesian updating procedure for Building 182 control room (λ_{CR}) fire frequency.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{CR} f_{A8} f_R A_2$$

where,

ϕ_{CM} - fire-induced core damage frequency for the Building 182 control room

A_2 is identical to those terms in previous sections

λ_{CR} - fire frequency for Building 182 control room

f_{A8} - area ratio of the PCS/SCS support systems control cabinet to that of all electrical cabinets in the control room

f_R - probability that the fire will not either be manually or automatically extinguished before damage occurs

Table 5.17 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.17

Building 182 Control Room Fire Scenario
Factors and Distribution

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{CR}	Gamma	1.7E-7	3.2E-3	7.4E-3
A_2	Lognormal		2.27E-3	(EF = 10)
f_{AB}	Maximum Entropy	0.17	0.34	0.68
f_R	Maximum Entropy	1.0E-3	0.01	0.10

5.10.8 Building 105, Room 9

Room 9 is located in the basement of Building 105. It contains a safety-related battery as well as conduit for the V4 valves which runs along the ceiling. Since this room is relatively small, a hot gas layer forms with the development of a large fire. This hot gas layer fails the V4 valves cabling interior to the conduit and thus fails ECCS (A). Random failure of the PCS/SCS (G) also occurs. Therefore, a transient (GAC) sequence is the consequence of a fire in this area in conjunction with other random failures.

Since a large fire anywhere in this room can form a hot gas layer no area ratio factor appears in the core damage equation. However, since only a large fire is capable of producing this hot gas layer, the severity ratio term (f_{S1}) is required.

The Q_R term in the core damage equation for room 9 represents the random failure of the PCS/SCS and is solved by the following equation:

$$Q_R = GSW2H33Q * T2H33R1 + FOPHPIPU.$$

These random failures represent either miscalibration of the HPIP DP instrumentation (FOHPIPU) or operator error to bypass 2H33 after a scram (GSW2H33Q, T2H33R1).

The frequency of fires in battery rooms (λ_{BAT}) was developed using battery room fire data in the first stage of the Bayesian updating procedure.

The term $Q_4(\tau_G)$ which assesses the probability of manual extinguishment of the fire before the hot gas layer forms was quantified in an identical fashion as the corresponding terms in previous sections.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{BAT} f_{S1} Q_4(\tau_G) Q_R$$

where,

ϕ_{CM} = fire-induced core damage frequency for Building 105, Room 9

$Q_4(\tau_G), f_{S1}$ are identical to those terms described in previous sections

Q_R is defined in the preceding discussion

λ_{BAT} = frequency of Room 9 fires

Table 5.18 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.18

Building 105, Room 9 Fire Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{BAT}	Gamma	1.6E-7	2.7E-3	6.1E-3
f_{S1}	Maximum Entropy	0.19	0.30	0.67
$Q_4(\tau_G)$	Maximum Entropy	0.83	0.99	1.0
Q_R	Lognormal		2.0E-4	(E.F. = 10)

5.10.9 Building 105, Room 8

Room 8 is adjacent to Room 9 in the Building 105 basement. The V4 valves cabling is in cable trays near the ceiling. Similarly, a large fire is required to form a hot gas layer which fails the V4 valves and thus the ECCS (A). As was the case for room 9 the same random failures lead to failure of the PCS/SCS (G). Therefore, a transient sequence (GAC) is the consequence of a fire in this area in conjunction with additional random failures.

Since Room 8 is a switchgear room its fire frequency (λ_{SGR}) was quantified in an identical fashion to those similar areas of Building 15.

All other core damage factors are identical to Room 9 except $Q_5(\tau_G)$ which has a slightly different probability distribution for time to manually extinguish the fire.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{SGR} f_{S1} Q_5(\tau_G) Q_R$$

where,

ϕ_{CM} = fire-induced core damage frequency for Building 105, Room 8

$Q_5(\tau_G)$, λ_{SGR} , f_{S1} , Q_R are identical to those terms described in previous sections

Table 5.19 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.19

Building 105, Room 8 Fire Scenario Factors and Distributions

Factor	Distribution	Lower Bound	Best Estimate	Upper Bound
λ_{SGR}	Gamma	6.2E-6	2.8E-3	5.9E-3
f_{S1}	Maximum Entropy	0.19	0.30	0.67
$Q_5(\tau_G)$	Maximum Entropy	0.70	0.99	1.0
Q_R	Lognormal		2.0E-4	(E.F. = 10)

5.10.10 Outlet Piping Area

The outlet piping area is located in the confinement portion of Building 105. As was the case for Rooms 8 and 9 fire damage in this area is to the V4 valve cabling which is enclosed in conduit. With the identical random failure of the PCS/SCS (G), a transient sequence (GAC) once again arises.

In this area the conduit was found to be approximately 9 ft from the floor and could be damaged by either a large or small fire. Thus, two severity ratio terms (f_{S1} and f_{S2}) appear in the core damage equation with their respective area ratio factors (f_{A9} and f_{A10}).

Fire frequency for this area within the confinement was developed utilizing generic nuclear containment fire data in the first Bayesian updating step. An additional area ratio factor (f_{A1D}) was required to ratio the fire frequency for the entire confinement area to that of the outlet piping area (fire zone 1D).

The term $Q_6(\tau_G)$ which assesses the probability of manual extinguishment of the fire before damage occurs to the V4 valve cabling was quantified in an identical fashion as the corresponding terms in previous sections. The same random failures (Q_R) failed the PCS/SCS as was the case for Rooms 8 and 9.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{conf} f_{A1D} Q_6(\tau_G) Q_R [f_{S1} f_{A9} + f_{S2} f_{A10}]$$

where,

ϕ_{CM} = fire-induced core damage frequency for the outlet piping area

$Q_6(\tau_G)$, f_{S1} , f_{S2} , Q_R are identical to those terms described in previous sections

f_{A1D} = area ratio of fire area 1D to that of the total confinement

λ_{conf} = frequency of confinement fires

f_{A9} = area ratio for a large fire within fire area 1D

f_{A10} = area ratio for a small fire within fire area 1D

Table 5.20 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

5.10.11 Building 105, Room 6

Room 6 is the N-Reactor cable spreading room located below the control room in Building 105. Three scenarios were postulated in this area. Two occurred in the west corners of the room where cabling came in for the V3 and V4 valves. Neither of these scenarios had a frequency of greater than 10^{-8} per year when credit was taken for the automatic halon system contained in this area. Although there was uncertainty in the exact cable routing for the V3 and V4 valves once inside room 6, area ratios would have had to be increased by factors of at least three to even survive screening. Since the overall fire-induced core damage frequency was found to be greater than 10^{-5} per year, exact cable routing information would have a negligible effect on the overall risk. Therefore, no further details will be discussed for either of them.

Table 5.20

Outlet Piping Area Fire Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{conf}	Gamma	2.1E-4	5.6E-3	0.011
f_{A1D}	Maximum Entropy	0.047	0.24	1.0
$Q_6(\tau_G)$	Maximum Entropy	0.40	0.99	1.0
Q_R	Lognormal		2.0E-4	(E.F. = 10)
f_{S1}	Maximum Entropy	0.19	0.30	0.67
f_{A9}	Maximum Entropy	0.019	0.096	0.48
f_{S2}	Maximum Entropy	0.33	0.70	0.81
f_{A10}	Maximum Entropy	0.01	0.051	0.26

The one remaining scenario had fire-related failures of all frontline cooling systems. Therefore, a transient sequence (GAC) arises due to fire damage alone. These cable trays come through the south wall from Corridor 1.

A detailed review was conducted on the automatic halon system in this area. A test had been conducted which maintained greater than six percent concentration of halon. In the plant walkdown it was judged that there was adequate fire detector coverage of the entire area. Therefore, the factor (Q_{HAL}) was based on generic system reliability data with no modification deemed appropriate.

The frequency of Room 6 fires (λ_{CSR}) was developed using cable spreading fire data in the first stage of the Bayesian updating procedure.

Only a large fire could damage the Flamastic-coated cables entering the room. Therefore, the severity factor (f_{S1}) along with an associated area ratio (f_{A11}) are included in the core damage equation.

The term $Q_7(\tau_G)$ gives credit for manual non-suppression of the fire before the automatic halon system would be actuated. It was quantified in an identical fashion as similar factors were in previous sections.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{CSR} Q_7(\tau_G) Q_{HAL} f_{A11} f_{S1}$$

where,

ϕ_{CM} = fire-induced core damage frequency for Room 6

$Q_7(\tau_G), f_{S1}$ are identical to terms of previous sections

Q_{HAL} = probability of automatic halon system failure

f_{A11} = area ratio for a large fire within Room 6

λ_{CSR} = frequency of Room 6 fires

Table 5.21 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.21

Building 105, Room 6 Fire Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{CSR}	Gamma	1.1E-3	7.5E-3	0.026
$Q_7(\tau_G)$	Maximum Entropy	0.48	0.99	1.0
Q_{HAL}	Lognormal	6.0E-3	0.06	0.20
f_{A11}	Maximum Entropy	1.0E-3	5.0E-3	0.025
f_{S1}	Maximum Entropy	0.19	0.30	0.67

5.10.12 Building 105, Corridor 1

Corridor 1 is immediately adjacent to Room 6 and borders it along its south wall. The cable runs for all frontline cooling systems pass into its south wall from Corridor 19 then across the corridor directly into

Room 6. Therefore, a fire can damage all frontline cooling systems leading to a transient sequence (GAC). For the base case analysis as part of plant modifications prior to restart, it was assumed that this area had complete coverage by a sprinkler system. The current plant configuration has very limited sprinkler coverage. As a sensitivity study the current plant configuration for corridor 1 was analyzed and had the effect of increasing this areas mean core damage frequency by approximately a factor of twenty. The term Q_{ws} gives credit for automatic extinguishment of the fire before damage occurs to the critical cabling.

The frequency of fires (λ_{105}) was developed using reactor building fire data in the first stage of the Bayesian updating procedure. An area ratio factor (f_{3B1}) was required to ratio this frequency for the building as a whole (only the ground level and basement floor area was counted in the calculation) to that of Corridor 1.

As was the case in Room 6 only a large fire was found to be capable of damaging the Flamastic-coated cabling. Therefore, a severity ratio factor (f_{S1}) along with its associated area ratio (f_{A12}) were utilized.

The term $Q_8(\tau_G)$ gives credit for manual extinguishment of the fire before damage occurs. It was quantified in an identical fashion as similar factors were in previous sections.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{105} f_{A12} f_{S1} f_{3B1} Q_8(\tau_G) Q_{ws}$$

where,

ϕ_{CM} = fire-induced core damage frequency for Corridor 1

$Q_8(\tau_G), f_{S1}$ are identical to terms of previous sections

λ_{105} = frequency of Building 105 fires

f_{B1} = area ratio of Corridor 1 to that of Building 105

f_{A12} = area ratio for a large fire within fire area 3B1

Q_{ws} = probability that the automatic suppression system does not suppress the fire before damage occurs.

Table 5.22 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

5.10.13 Building 109, Corridor 19 and 109 Basement

The same cable runs that were found in Corridor 1 and Room 6 traverse through Corridor 19 then into the Building 109 Turbine Annex basement and exit underground along the south wall of the 109 basement. A fire can once again damage all frontline cooling systems leading to a transient sequence (GAC). Cable trays in this area are three across and four high.

Table 5.22

Building 105, Corridor 1 Fire Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{105}	Gamma	0.011	0.02	0.033
f_{3B1}	Maximum Entropy	2.0E-3	0.01	0.05
$Q_B(\tau_G)$	Maximum Entropy	0.40	0.99	1.0
f_{A12}	Maximum Entropy	0.016	0.082	0.41
f_{S1}	Maximum Entropy	0.19	0.30	0.67
Q_{WS}	Maximum Entropy	6.3E-3	0.038	0.047

It is known from cable tracing some of the specific tray locations for the frontline cooling systems. From this limited tracing, some of the bottom two vertical trays have cabling that once engulfed in a fire could fail the PCS/SCS (G), ECCS (A), and GSCS (C). It was, therefore, conservatively assumed that any of the three horizontal bottom trays and the cable tray immediately above any of them contained the requisite cabling to lead to a GAC sequence. However, for most of the area ratio (f_{19}) the fire actually engulfed four cable trays. These trays were two adjacent horizontal bottom trays and the two cable trays immediately above them. Credit was given for the Flamastic-coating found on the cabling in this area. It was assumed that the fire could not spread along the cable trays and, therefore, had to be very close to the cabling to cause damage.

An automatic sprinkler system is located in this area. Actuation for this system is through a fusible link activated by heat from the fire for each individual sprinkler head. Sprinkler heads were found to be centered below the middle tray approximately 10 ft apart. These sprinklers spray downwards toward the floor. Also, additional sprinklers were noted near the ceiling with approximately 20 ft of separation. Since self-ignited cable fires were dismissed due to the Flamastic-coating on the cables, the fire source would have to be a transient

combustible burning upwards from the floor. Credit was given for the sprinkler system below the cable trays. After inspection by a fire testing expert, it was assumed that if any point of the fire was within 1 meter radially of a sprinkler head, that the sprinkler would activate and suppress the fire before any cabling would be damaged. This led to an overall reduction of the area ratio factor of approximately 50 percent. No credit was given for the sprinkler heads above the cabling due to its wider sprinkler head separation and the blocking effect the Flamastic-coated cable trays would have on heat from the fire below.

For the base case analysis, as part of plant modifications prior to restart, it was assumed that this area had complete coverage by a sprinkler system. As a sensitivity study the current plant configuration as described in the previous paragraph for corridor 19 was analyzed and had the effect of increasing this areas mean core damage frequency by approximately a factor of ten. The term Q_{WS} gives credit for automatic extinguishment of the fire before damage occurs to the critical cabling.

Since only a large fire was of sufficient size to engulf the bottom two cable trays, the severity ratio factor (f_{S1}) was utilized in the core damage equation.

Credit was given for manual extinguishment of the fire before a sprinkler was activated through the term $Q_g(\tau_g)$. This term was quantified in an identical fashion to similar terms in the previous section.

The frequency of Building 109 fires (λ_{109}) was developed using turbine building data in the first stage of the Bayesian updating procedure.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{109} f_{19} f_{S1} Q_g(\tau_g) Q_{WS}$$

where,

ϕ_{CM} = fire-induced core damage frequency for Building 109 basement and Corridor 19

$Q_g(\tau_g), Q_{WS}, \lambda_{109}, f_{S1}$ are identical to terms of previous sections

f_{19} = area ratio of Building 109 basement and corridor to that of the total of Building 109

Table 5.23 gives the values of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.23

Building 109, Basement and Corridor 19 Fire Scenario
Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{109}	Gamma	0.022	0.043	0.11
f_{19}	Maximum Entropy	4.0E-3	0.02	0.10
$Q_9(r_G)$	Maximum Entropy	0.43	0.99	1.0
f_{S1}	Maximum Entropy	0.19	0.30	0.67
Q_{WS}	Maximum Entropy	6.3E-3	0.038	0.047

5.10.14 Main Control Room

The Building 105 main control room is located on the ground floor directly above Room 6. One scenario based on a transient sequence (GAC) remained after screening.

This scenario assumed abandonment of the control room due to smoke from a cabinet fire. Credit was given for quick extinguishment of the fire since the control room is continually manned and contains an automatic halon suppression system. Control room operators have access to Scott air packs and are trained in their use. Therefore, some credit was given for manual extinguishment of the fire before the automatic halon system could actuate.

The current control room abandonment procedure (due to fire) does not direct operators to man the remote shutdown panel (which is not yet fully operational). This current configuration was analyzed as a sensitivity study and had the effect of increasing control room (fire-induced) core damage frequency by approximately a factor of five.

The base case configuration assumed a fully operational remote shutdown panel with procedures in place to man it immediately in the event of smoke-induced abandonment of the control room. The panel in which the fire originates is a critical consideration to the timing of core damage. Consistent with the internal events analysis if damage was sustained to control functions of either PCS/SCS or ECS systems core damage was assumed to occur in approximately seven minutes. These cabinets

comprised 20 percent of the overall cabinet area in the control room and are represented by the term f_R in the core damage equation. No credit for plant recovery was allowed due to the short amount of time of the scenario.

A second scenario postulated a fire in any other cabinet in the control room. Discussions with operations personnel indicated that the plant should stay in a stable configuration for a best estimate time of 30 minutes. In this case recovery was allowed from the remote shutdown panel. The term Q_{R1} in the core damage equation gives credit for operators going to once-through cooling in this 30-minute time frame.

The frequency of control room fire (λ_{CR}) was developed in an identical fashion to that of Section 5.10.7.

Thus, the core damage equation is as follows:

$$\phi_{CM} = \lambda_{CR} f_R [f_A + (1-f_A)Q_{R1}]$$

where,

ϕ_{CM} = fire-induced core damage frequency for the main control room.

λ_{CR} , f_R are identical to these terms in Section 5.10.7

f_A = that percentage of cabinet area in the control room containing PCS/SCS and ECS functions

Q_{R1} = probability of non-recovery from the remote shutdown panel

Table 5.24 gives the value of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.24

Main Control Room Fire Scenario Factors and Distributions

<u>Factor</u>	<u>Distribution</u>	<u>Lower Bound</u>	<u>Best Estimate</u>	<u>Upper Bound</u>
λ_{CR}	Gamma	1.7E-7	3.2E-3	7.4E-3
f_R	Maximum Entropy	1.0E-3	0.01	0.10
f_A	Maximum Entropy	0.10	0.20	0.40
Q_{R1}	Maximum Entropy	1.0E-3	0.01	0.10

5.10.15 Building 151 Switchgear

Building 151 contains Bus "A" 13.8 kV switchgear, including the switchgear cubicles for Circulating Raw Water Pumps #1 and #3. Also included are the main breaker for the "A" bus offsite power supply, station service transformer breaker, and the 13.8 kV Bus "B" tie breaker.

This scenario results in transient sequence $GA_2\bar{C}$, as described in previous Building 184 scenarios. A fire-induced failure of the switchgear in this building results in a loss of offsite power which leads to instrument air failure. This fails PCS/SCS (G) which, along with ECCS (A_2) random failure and the GSCS succeeding (C), yields the accident sequence.

The area ratio factor (f_{A13}) is developed based upon the ratio of offsite power breaker cubicle area to that of the total switchgear cabinet area.

The frequency of switchgear room fires (λ_{SGR}) was developed in an identical fashion to that summarized in Section 5.10.3.

Finally, the term $Q_{10}(\tau_G)$ which assesses the probability of manual extinguishment before fire damage occurs is quantified in an identical fashion as the corresponding terms in previous sections.

The core damage equation is as follows:

$$\phi_{cm} = \lambda_{SGR} f_{A13} Q_{10}(\tau_G) A_2$$

where,

ϕ_{cm} - fire-induced core damage frequency for Building 151 main floor

$Q_{10}(G), A_2$ are identical to those terms described in previous sections

λ_{SGR} - frequency of Building 151 main floor switchgear fires

f_{A13} - area ratio of the total switchgear cabinet area to that of the total Building 151 floor area

Table 5.25 gives the value of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.25

Building 151 Switchgear Fire Scenario
Factors and Distributions

Factor	Distribution	Lower Bound	Best Estimate	Upper Bound
λ_{SGR}	Gamma	6.2E-6	2.8E-3	5.9E-3
f_{A13}	Maximum Entropy	0.025	0.13	0.63
$Q_{10}(r_G)$	Maximum Entropy	0.4	0.99	1.0
A_2	Lognormal		2.27E-3	(E.F.=10)

5.10.16 Building 184 Control Room

Building 184 contains the instrument and service air compressors, a Building 184 control room, and power cabling and controls associated with the compressors.

Fire damage in the control room results in loss of offsite power and fails the instrument air system. Failure of instrument air fails the PCS/SCS and leads to a GA₂C sequence as described in Sections 5.10.3 through 5.10.5.

Thus, the core damage equation is as follows:

$$\phi_{cm} = \lambda_{CR} f_R f_{A14} A_2$$

where,

ϕ_{cm} = fire-induced core damage frequency for the Building 184 control room

λ_{CR} , f_R , A_2 are identical to those terms in previous sections

f_{A14} = area ratio of the control cabinets associated with offsite power to that of all the electrical cabinets in the Building 184 control room.

Table 5.26 gives the value of each of these factors as well as their associated distribution and upper and lower bounds.

Table 5.26

Building 184 Control Room Fire Scenario
Factors and Distributions

Factor	Distribution	Lower Bound	Best Estimate	Upper Bound
λ_{CR}	Gamma	1.7E-7	3.2E-3	7.4E-3
f_R	Maximum Entropy	1.0E-3	0.01	0.10
f_{A14}	Maximum Entropy	0.054	0.27	1.0
A_2	Lognormal		2.27E-3	(E.F.=10)

5.11 Conclusions

A detailed screening analysis found that most plant areas had a negligible contribution to fire-induced core damage frequency. The following fifteen areas had core damage contributions of greater than 10^{-8} per year.

- a. Cell Access Corridor
- b. Building 153 Main Floor
"B" Switchgear
- c. Building 153 Main Floor
"A" Switchgear
- d. Building 184 Instrument Air Compressor
Cable Runs
- e. Building 184 Instrument Room
Compressor Area
- f. Building 182 Control Room
- g. Building 105, Room 9
- h. Building 105, Room 8
- i. Outlet Piping Area
- j. Building 105, Room 6
- k. Building 105, Corridor 1
- l. Building 109, Corridor 19 and Basement
- m. Main Control Room
- n. Building 151 Switchgear
- o. Building 184 Control Room

The overall total (mean) core damage frequency was found to be $1.96E-5$ per year.

Detailed tracing of control and power cabling revealed that fires could damage the most plant safety functions in cable "pinchpoint" areas.

For specific fire scenario contributions to core damage frequency, refer to Table 5.1. The event sequences each of these scenarios mapped into are given in Table 5.2.

The TEMAC code produces the following three importance/uncertainty measures:

- a. Risk reduction--For each basic event (either fire-related or random factor), its probability is set to zero and the reduction in core damage frequency is calculated.
- b. Risk increase--For each basic event its probability is set to 1.0 and the increase in core damage frequency is calculated.
- c. Uncertainty importance--For each basic event, its distribution is eliminated from the overall uncertainty calculation by setting the event to its mean value and the percent decrease in the logarithm of the overall uncertainty is then calculated.

The three most dominant basic events from a risk-reduction standpoint (listed in order of importance) are:

- a. f_{S1}
- b. Q_{WS}
- c. f_{19}

All three of these terms are defined in the core damage equation for corridor 19. Both terms a and b appear in the dominant cut set as well as other less dominant cut sets and are, therefore, the most important factors to risk reduction. Term c appears only in the dominant cut set and is therefore ranked slightly lower.

The three most dominant basic events from a risk-increase standpoint (listed in order of importance) are:

- a. Q_R = random failure of PCS/SCS
- b. A_2 = random failure of one check valve and thus failure to cool one-sixteenth of the core.
- c. f_R = probability for any of the three control rooms that a fire will not be manually extinguished before damage occurs.

Q_R appears in the core damage equations for both rooms 8 and 9. It is a relatively low probability event ($2.0E-4$) and therefore is the most dominant factor in risk increase. A_2 appears in many core damage cut sets and is also relatively low in probability ($2.3E-3$). The term f_R appears in all control room scenarios but has a higher probability than either of the preceding terms.

The three most dominant basic events from an uncertainty importance standpoint (listed in order of importance) are:

- a. f_{19}
- b. A_2
- c. f_{S1}

F_{19} is the most dominant term because it appears in the most dominant cut set and its uncertainty is relatively high. A_2 and f_{S1} both appear in a great many less dominant cut sets and also have a relatively high uncertainty.

As a result of the application of these methods to the N-Reactor plant and previously to the Peach Bottom and Surry plants in the NUREG-1150 analyses, fire analyses can now be performed to an equal level of detail when compared to an internal events analyses. The core damage frequency estimates can be compared directly to the internal event results, and one integrated analysis combining the results of the internal event analysis with the fire and other external events analyses can be accomplished. This integrated analysis gives a means of directly comparing the importance of events regardless of which analysis/analyses they occur in.

5.12 References

1. Seabrook Station Probabilistic Safety Assessment, Section 9.4, Public Service Company of New Hampshire and Yankee Atomic Electric Company, December 1983.
2. Severe Accident Risk Assessment Limerick Generating Station, Chapter 4, Main Report, Philadelphia Electric Company, Report #4161, April 1983.
3. J. A. Lambright, S. P. Nowlen, V. F. Nicolette, and M. P. Bohn, Fire Risk Scoping Study: Current Perception of Unaddressed Fire Risk Issues, Sandia National Laboratories, Albuquerque, NM, SAND88-0177, NUREG/CR-5088, December 1988.
4. W. T. Wheelis, Users Guide for a Personal-Computer-Based Nuclear Power Plant Fire Data Base, Sandia National Laboratories, Albuquerque, NM, SAND86-0300, NUREG-CR-4586, August 1986.
5. R. L. Iman, S. C. Nora, Modeling Time to Recovery and Initiating Event Frequency for Loss of Off-Site Power Incidents at Nuclear Power Plants, Sandia National Laboratories, Albuquerque, NM, SAND87-2428, NUREG/CR-5032, January 1988.
6. R. B. Worrell, SETS Reference Manual, Sandia National Laboratories, Albuquerque, NM, SAND83-2675, NUREG/CR-4213, May 1985.
7. D. W. Stack, A SETS User's Manual for Accident Sequence Analysis, Sandia National Laboratories, Albuquerque, NM, SAND83-2238, NUREG/CR-3547, January 1984.
8. N. O. Siu, COMPBRN - A Computer Code for Modeling Compartment Fires, University of California, UCLA-ENG-8257, NUREG/CR-3239, May 1983.
9. V. Ho, N. O. Siu, G. Apostolakis, "COMPBRN III - A Computer Code for Modeling Compartment Fires," University of California, UCLA-ENG-8524, November 1985.
10. C. Ruger, J. L. Boccio, and M. A. Azarm, Evaluation of Current Methodology Employed in Probabilistic Risk Assessment (PRA) of Fire Events at Nuclear Power Plants, Brookhaven National Laboratory, NUREG/CR-4229, May 1985.
11. V. F. Nicolette, S. P. Nowlen, J. A. Lambright, "Observations Concerning the COMPBRN III Fire Growth Code," Sandia National Laboratories, Albuquerque, NM, SAND88-2160C, April 1989.

12. W. T. Wheelis, "Transient Combustible Fuel Sources Found at Nuclear Power Plants (Data)," Letter Report, Sandia National Laboratories, Albuquerque, NM, July 1984.
13. V. F. Nicolette, and S. P. Nowlen, "A Critical Look at Nuclear Qualified Electrical Cable Insulation Ignition and Damage Thresholds," SAND88-2161, paper accepted for presentation at Operability of Nuclear Systems in Normal and Adverse Environments, Lyon, France, September 18-22, 1989.
14. J. A. Lambright, Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program, Fire Analysis, Volume 9, SAND87-7157, NUREG/CR-4832, 1990.
15. R. L. Iman, M. J. Shortencarrier, A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models, Sandia National Laboratories, Albuquerque, NM, SAND83-2365, NUREG/CR-3024, March 1984.
16. R. L. Iman, M. J. Shortencarrier, A User's Guide for the Top Event Matrix Analyses Code (TEMAC), Sandia National Laboratories, Albuquerque, NM, SAND86-0960, NUREG/CR-4598, August 1986.

APPENDIX A
MANUAL FIRE SUPPRESSION ANALYSIS

An important component in determination of the frequency of fire-induced core damage scenarios is the ability of the plant fire brigade to respond to and extinguish fires in a timely fashion before damage can occur to plant systems and components important to safety. The COMPBRN fire propagation code predicts the time to ignition or damage of critical cables and components. A thorough walkdown of the plant and review of its fire brigade practices is also necessary to determine the probability of manual suppression in a given time frame for all critical plant areas. The COMPBRN predicted fire-induced equipment or cable damage times are then used in conjunction with the distribution curves on time to suppression of fires to obtain the probability that a given fire will damage critical safety equipment before it can be suppressed. This Appendix is a summary of the analysis methods used and the results of the analysis for the N-Reactor plant.

During the initial plant visit, a thorough walkdown of the plant was conducted to assess plant fire fighting capabilities and to note access routes for extinguishing fires in the critical plant locations. Information was gathered regarding installed automatic fire suppression systems; manual fire fighting equipment such as hose reels and portable extinguishers; fire detectors and alarm provisions; limitations to access (locked doors, security restrictions, etc.); distances from external points of access to the fire locations; and room size and geometry as they relate to access and the level of smoke obscuration. Interviews of the plant fire department and a comprehensive tour of the fire department facilities were conducted to assess fire department response time, distances to fire locations, training, and equipment provisions.

In establishing the time from established burning (flame approximately 10 in. tall) to suppression, the information gathered during the plant visit was used to determine the time to reach a number of discrete phases or events. Those phases included the following:

1. Established burning to detection.
2. Detection to alarm.
3. Fire Department site response.
4. Arrival at the room of origin.
5. Finding the fire.
6. Agent application.
7. Extinguishment or substantial control of the fire.

For the estimated times, maximum, minimum, and what was considered to be the best estimate or median time, were provided. Tables A-1 to A-23 summarize these estimates of cumulative times from established burning to the end of each phase. Also provided in addition to the room and location is

information on the type of detection, if present; the type of automatic suppression; and notes on assessment of human detection, manual suppression accessibility, and equipment available.

The degree of confidence in the estimated times varies, depending on the building features and the phase under consideration. For Phase 1, the confidence in the time estimates is good where automatic detectors are provided. For human detection, there is less confidence in the time estimates. However, among the areas inspected during the plant visit, there are only two areas without automatic fire detection.

For Phase 2, the level of confidence is good to excellent for spaces equipped with automatic detection, integrated with the fire alarm panel. The level of confidence is fair for human detection.

In Phase 3, the level of confidence is good to excellent. According to meetings with Hanford Fire Department personnel, the distance between Fire Station 100 to the N-Reactor is about 3 miles. Based on these discussions, it would take 3.5 minutes for five fire fighters and one Captain to reach the N-Reactor. Presumably, this assumes that all fire fighters are within the station and ready to move upon receipt of an alarm. For the worst case, this evaluation assumes that it will take 3 additional minutes for fire fighters to gather from different parts of the site and get ready to respond to an alarm.

The level of confidence for Phase 4 is good. It is assumed that trained occupants have already found the room of origin and are waiting to lead the fire fighters to the room. Time required to arrive at the fire location has a strong dependence on the distance between the plant area and the exterior hose outlet, e.g. building entrance. For distances up to 150 ft, the time required to arrive at the fire location is at least 2 minutes and at most about 6 minutes. For distances between 150 and 250 ft, these minimum and maximum times are on the order of 4 and 11 minutes.

Regarding Phase 5, it is presumed that the level of smoke obscuration outside the room of fire origin is minimal. For example, smoke movement throughout the building by means of the HVAC system is not taken into account. Time to locate the fire depends on several factors; the size of the area of origin, the size of the fire, the amount of equipment within the fire area, the level of smoke obscuration, etc. In general, it should be easier to locate the fire within a large room which contains a limited amount of equipment, than within a small room with more equipment.

Finally, the level of confidence for Phase 6 and Phase 7 is good. A fire discovered early may be extinguished by means of readily available fire extinguishers. However, occupant fire extinguishment is not included in this analysis.

A computer model was used to estimate the fire size and the depth and temperature of smoke in the room. This information was utilized in predicting conditions within the room at the time of detection and fire department arrival at the room, and for estimating the time to find and extinguish/suppress the fire. Room geometry and heat release rates are inputs for this computer model. A second computer model was used to predict the time to actuation of automatic detectors, and/or automatic sprinklers.

The total cumulative times to extinguishment of Phase 7 were then entered into a Sandia-developed statistical analysis and plotting routine which includes a curve fitting parameter estimation routine. The results, depicted in Figures A-1 through A-21, are user defined log-normal plots of suppression probability versus time to suppression for the various plant areas. The objective of the curve fitting routine is to establish a "best fit" log normal probability distribution function for each of the data sets of time to suppression (Phase 7 cumulative times) versus cumulative probability of suppression.

Due to the limited number of data points (minimum, median, and maximum), the curve fit is in some cases difficult to establish and still result in confidence in the correlation, especially at the lower end of the curve. Since the times to damage determined by the COMPBRN code are typically in the range of 4 to 12 minutes, the lower end of the curve was deemed the most important with respect to establishing the best fit. Therefore, only two points, the minimum and the median, were entered into the plotting routine to yield curves constrained to pass through these lower values. The resulting curves are judged to be more representative of the suppression probabilities in the range of interest.

Once the suppression time plots were generated, the probability of not being able to suppress the fire in a given time frame in any plant area was determined. Uncertainties in this estimate were generated by adding and subtracting 15 minutes from the COMPBRN predicted time to damage. This time difference from the COMPBRN predicted results could be due to a number of physical uncertainties including prolonged smoldering of a fire prior to ignition and is felt to represent realistic bounds on time to damage.

Table A-1

N-Reactor Manual Fire Suppression
Building 105

Room: Cable Spreading Room, Fire Area: 3A

AUTOMATIC DETECTION: Smoke detectors within cabinets and at the ceiling.

AUTOMATIC SUPPRESSION: Total Flooding Halon system.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Excellent hydrant accessibility.

EQUIPMENT AVAILABLE: - Fire extinguishers.
- Manual CO₂ system with hose on reels.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	2.0*	20.0**	5.0*
2. Alarm	2.0	21.0	5.0
3. Fire Department Site Response	5.5	28.0	10.0
4. Arrival at the Room of Origin	7.5	34.0	14.0
5. Finding the Fire	8.0	37.0***	15.0
6. Agent Application	8.0	38.0	16.0
7. Extinguishment	9.0	45.0	20.0

NOTES: * Successful control or extinguishment of the fire by means of the total flooding Halon system is neglected. Successful operation of the Halon system will limit the fire size at about 30 seconds after agent discharge.

** At the time of detection, the fire size is beyond the capability of fire extinguishers.

*** At the time that the fire department finds the fire, the room is close to full room involvement with room temperature exceeding 1,000°F.

Table A-2

N-Reactor Manual Fire Suppression
 Building 105
 Room: 9 Battery Room, El. -16.0, Fire Area: 3B

AUTOMATIC DETECTION: Smoke detector.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility.

EQUIPMENT AVAILABLE: Fire extinguisher in room 3b-8.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	2.5	6.0	4.0
2. Alarm	2.5	7.0	4.5
3. Fire Department Site Response	6.0	13.5*	9.5*
4. Arrival at the Room of Origin	11.0**	25.5	17.0
5. Finding the Fire	12.0	27.0	18.0
6. Agent Application	12.0	28.0	18.0
7. Extinguishment	15.0	34.0	22.0

NOTES: * At the time of fire department site response, the room may be fully involved in fire with room temperature exceeding 1,200°F.

** At the time of fire department arrival at the room of origin, the room may be fully involved in fire.

Table A-3

N-Reactor Manual Fire Suppression

Building 105

Room: 8 Switchgear Room, El. -16.0, Fire Area: 3B

AUTOMATIC DETECTION: Smoke Detector.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility.

EQUIPMENT AVAILABLE: Fire extinguishers outside the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	2.0	10.0	5.5
2. Alarm	2.0	11.0	6.0
3. Fire Department Site Response	5.5	17.5*	11.0
4. Arrival at the Room of Origin	10.5	29.5	19.5**
5. Finding the Fire	10.5	31.0	21.0
6. Agent Application	11.0	32.0	22.0
7. Extinguishment	12.0	38.0	27.0

NOTES: * At the time of fire department site response, the room may be fully involved in fire with room temperature exceeding 1,200°F.

** At the time of fire department arrival at the room of origin, the room may be fully involved in fire.

Table A-4
N-Reactor Manual Fire Suppression
Building 105
Room: 2 and 47, El. -16.0, Fire Area: 3B

AUTOMATIC DETECTION: Automatic sprinklers.

AUTOMATIC SUPPRESSION: Automatic sprinklers (ceiling mounted sprinklers in Section 3b and sidewall sprinklers in the East 3b).

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility.

EQUIPMENT AVAILABLE: H₂O fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	7.0*	18.0**	12.0**
2. Alarm	8.0	19.0	13.0
3. Fire Department Site Response	11.5	25.5***	18.0
4. Arrival at the Room of Origin	15.5	34.5	24.0****
5. Finding the Fire	16.0	36.0	25.0
6. Agent Application	16.0	38.0	26.0
7. Extinguishment	17.0	43.0	28.0

NOTES: * Successful control or extinguishment of the fire by means of automatic sprinklers is neglected. Successful operation of automatic sprinklers would limit the fire size at about 1.0 minute after agent discharge (approximately 8.0 minutes).

** At the time of detection, the fire size is beyond the capability of fire extinguishers.

*** At the time of fire department site response, the room is fully involved in fire with room temperature exceeding 1,200°F.

**** At the time of fire department arrival at the room of origin, the room is fully involved in fire with room temperature exceeding 1,200°F.

Table A-5

N-Reactor Manual Fire Suppression

Building 105

Room: Corridor 1, El. -16.0, Fire Area: 3B1

AUTOMATIC DETECTION: Automatic sprinklers.

AUTOMATIC SUPPRESSION: Automatic sprinklers.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility.

EQUIPMENT AVAILABLE: 2-1/2 gallon pressurized water extinguisher.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0	11.0**	7.0*
2. Alarm	5.0	12.0	8.0
3. Fire Department Site Response	8.5	18.5	13.0***
4. Arrival at the Room of Origin	12.5****	27.5	19.0
5. Finding the Fire	13.0	28.0	19.5
6. Agent Application	13.0	28.0	19.5
7. Extinguishment	18.0	38.0	27.0

NOTES: * Successful control or extinguishment of the fire by means of automatic sprinklers is neglected. Successful operation of sprinklers would limit the fire size at about 1.0 minute after agent discharge (approximately 8.0 minutes).

** At the time of detection, the room is fully involved in fire with room temperature exceeding 1,000°F.

*** At the time of fire department site response, the room is fully involved in fire

**** At the time of fire department arrival at the room of origin, the room is fully involved in fire.

Table A-6
N-Reactor Manual Fire Suppression
Building 105
Room: Room 30, El. -16.0, Fire Area: 3G

AUTOMATIC DETECTION: Rate of rise heat detectors.
AUTOMATIC SUPPRESSION: Deluge system for equipment.
HUMAN DETECTION: Occasional.
MANUAL SUPPRESSION: Good hydrant accessibility.
EQUIPMENT AVAILABLE: Fire extinguishers

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	2.0*	6.0	4.0**
2. Alarm	2.0	7.0	4.0
3. Fire Department Site Response	5.5	13.5***	9.0***
4. Arrival at the Room of Origin	10.0****	24.5	16.0
5. Finding the Fire	12.0	28.0	18.0
6. Agent Application	13.0	30.0	20.0
7. Extinguishment	15.0	35.0	24.0

NOTES: * Successful control or extinguishment of the fire by means of the deluge system is neglected. Successful operation of the deluge system would limit the fire size at about 1.0 minute after agent discharge (approximately 3.0 minutes).

 ** At the time of detection, the fire size is beyond the capability of fire extinguishers.

 *** At the time of fire department site response, the room is fully involved in fire with room temperature exceeding 1,000°F.

 **** At the time of fire department arrival at the room of origin, the room is fully involved in fire with temperature exceeding 1,000°F.

Table A-7

N-Reactor Manual Fire Suppression
 Building 105
 Room: Corridor 5, El. -16.0, Fire Area: 3J

AUTOMATIC DETECTION: Heat detectors above cable trays.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility.

EQUIPMENT AVAILABLE: H₂O fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	5.0	15.0	7.5
2. Alarm	6.0	16.0	7.5
3. Fire Department Site Response	9.5	22.5*	12.5
4. Arrival at the Room of Origin	13.5	31.5	18.5
5. Finding the Fire	14.0	32.5	19.5**
6. Agent Application	14.0	33.0	20.0
7. Extinguishment	15.0	35.0	22.0

NOTES: * At the time of fire department site response, the room is fully involved in fire with room temperature exceeding 1,200°F.

** At the time that the fire department finds the fire, the room is fully involved in fire with room temperature exceeding 1,200°F.

Table A-8

N-Reactor Manual Fire Suppression
 Building 105
 Room: Main Control Room, Fire Area: 4A

AUTOMATIC DETECTION: Smoke detectors.

AUTOMATIC SUPPRESSION: Total flooding Halon system.

HUMAN DETECTION: Excellent, continuously occupied.

MANUAL SUPPRESSION: Excellent hydrant accessibility.

EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	2.5*	20.0	3.5*
2. Alarm	2.5	21.0	3.5
3. Fire Department Site Response	6.0	27.5	8.5
4. Arrival at the Room of Origin	8.0	33.5	12.5
5. Finding the Fire	8.0	34.0	12.5
6. Agent Application	8.5	35.0	13.5
7. Extinguishment	9.5	38.0	14.0

NOTES: * Successful control or extinguishment of the fire by means of the total flooding Halon system is neglected. Successful operation of the Halon system will limit the fire size at about 30 seconds after agent discharge.

Table A-9
N-Reactor Manual Fire Suppression
Building 109
Room: Corridor 19, El. -16.0, Fire Area: 21

AUTOMATIC DETECTION: Automatic sprinklers below cable trays.
AUTOMATIC SUPPRESSION: Automatic sprinklers below cable trays.
HUMAN DETECTION: Occasional.
MANUAL SUPPRESSION: Excellent hydrant accessibility.
EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	3.0*	10.0**	5.0
2. Alarm	3.0	11.0	5.0
3. Fire Department Site Response	6.5***	18.0	10.0****
4. Arrival at the Room of Origin	10.5	27.0	16.0
5. Finding the Fire	12.0	30.0	18.0
6. Agent Application	12.5	31.0	19.0
7. Extinguishment	14.0	40.0	23.0

NOTES: * Successful control or extinguishment of the fire by means of automatic sprinklers is neglected. Successful operation of sprinklers will limit the fire size at about 1.0 minute after agent discharge (approximately 4.0 minutes).

 ** At the time of detection, the area is fully involved in fire with room temperature exceeding 1,200°F.

 *** At the time of fire department site response, the room is close to full room involvement situation with room temperature exceeding 1,000°F.

 **** At the time of fire department site response, the area is fully involved in fire with room temperature exceeding 1,200°F.

Table A-10

N-Reactor Manual Fire Suppression
 Building 109
 Room: El. -16.0 Basement, Fire Area: 21

AUTOMATIC DETECTION: Automatic sprinklers.

AUTOMATIC SUPPRESSION: Automatic sprinklers. Some sprinklers are blocked by equipment.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility. 1-1/2" standpipes with no hose.

EQUIPMENT AVAILABLE: Fire extinguishers. The size of fire extinguishers does not appear to be adequate for anticipated fires.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	6.0**	40.0*	20.0*
2. Alarm	6.0	42.0	21.0
3. Fire Department Site Response	9.5	48.5	26.0
4. Arrival at the Room of Origin	13.5	58.0	32.0
5. Finding the Fire	14.0	60.0	33.0
6. Agent Application	14.0	61.0	33.0
7. Extinguishment	16.0	70.0	38.0

NOTES: * At the time of detection, the fire size is beyond the capability of fire extinguishers.

** Successful control or extinguishment of the fire by means of automatic sprinklers is neglected. Successful operation of sprinklers would limit the fire size at about 2.0 minutes after agent discharge (approximately 8.0 minutes).

Table A-11

N-Reactor Manual Fire Suppression
Building 109

Room: El. 0.0 Turbine Floor, Fire Area: 21

AUTOMATIC DETECTION: None.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility. 1-1/2" hose stations spaced evenly throughout the area.

EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	15.0	90.0	50.0
2. Alarm	16.0	92.0	52.0
3. Fire Department Site Response	19.5	99.0	57.0
4. Arrival at the Room of Origin	22.0	105.0	61.0
5. Finding the Fire	23.0	112.0	65.0
6. Agent Application	23.0	115.0	66.0
7. Extinguishment	24.0	120.0	68.0

Table A-12

N-Reactor Manual Fire Suppression
 Building 153
 Room: El. -20.0, Fire Area: 26A

AUTOMATIC DETECTION: Three rate of rise heat detectors.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Excellent hydrant accessibility.

EQUIPMENT AVAILABLE: Fire extinguishers outside the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0	11.5*	6.0**
2. Alarm	5.0	12.5	6.0
3. Fire Department Site Response	8.5	19.0***	11.0
4. Arrival at the Room of Origin	10.5	25.0	15.0***
5. Finding the Fire	11.0****	26.0	16.0
6. Agent Application	13.0	30.0	19.0
7. Extinguishment	20.0	40.0	28.0

NOTES: * At the time of detection, the room is close to full room involvement, with room temperature exceeding 1,000°F.

** At the time of detection, the fire size is beyond the capability of fire extinguishers.

*** At this time, the room is fully involved in fire with room temperature exceeding 1,200°F.

**** At the time that the fire is found, the room is close to full room involvement, with room temperature exceeding 1,000°F.

Table A-13
N-Reactor Manual Fire Suppression
Building 153
Room: El. -12.0, Fire Area: 26B

AUTOMATIC DETECTION: Three rate of rise heat detectors.
AUTOMATIC SUPPRESSION: None.
HUMAN DETECTION: Occasional.
MANUAL SUPPRESSION: Excellent hydrant accessibility.
EQUIPMENT AVAILABLE: Fire extinguishers outside the room.
EVENT/PHASE DESCRIPTION

	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0	11.5*	6.0**
2. Alarm	5.0	12.5	6.0
3. Fire Department Site Response	8.5	19.0***	11.0
4. Arrival at the Room of Origin	10.5	25.0	15.0***
5. Finding the Fire	11.0****	26.0	16.0
6. Agent Application	13.0	30.0	19.0
7. Extinguishment	20.0	40.0	28.0

NOTES: * At the time of detection, the room is close to full room involvement, with room temperature exceeding 1,000°F.
 ** At the time of detection, the fire size is beyond the capability of fire extinguishers.
 *** At this time, the room is fully involved in fire with room temperature exceeding 1,200°F.
 **** At the time that the fire is found, the room is close to full room involvement, with room temperature exceeding 1,000°F.

Table A-14

N-Reactor Manual Fire Suppression
 Building 153
 Room: El. 0.0, Fire Area: 27A

AUTOMATIC DETECTION: Three rate of rise heat detectors, 20" below ceiling.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility.

EQUIPMENT AVAILABLE: Fire extinguishers within the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0	9.0*	6.0*
2. Alarm	5.0	9.0	6.0
3. Fire Department Site Response	8.5	15.5	11.0
4. Arrival at the Room of Origin	10.5	21.0	15.0
5. Finding the Fire	11.0	23.0	16.5
6. Agent Application	11.0	24.0	17.0
7. Extinguishment	12.0	25.0	18.0

NOTES: * At the time of detection, the fire size is beyond the capability of fire extinguishers.

Table A-15

N-Reactor Manual Fire Suppression
 Building 153
 Room: El. 0.0, Fire Area: 27B

AUTOMATIC DETECTION: Three rate of rise heat detectors, 20" below ceiling.

AUTOMATIC SUPPRESSION: None.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Very good hydrant accessibility.

EQUIPMENT AVAILABLE: Fire extinguishers within the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0	9.0*	6.0*
2. Alarm	5.0	9.0	6.0
3. Fire Department Site Response	8.5	15.5	11.0
4. Arrival at the Room of Origin	10.5	21.0	15.0
5. Finding the Fire	11.0	23.0	16.5
6. Agent Application	11.0	24.0	17.0
7. Extinguishment	12.0	25.0	18.0

NOTES: * At the time of detection, the fire size is beyond the capability of fire extinguishers.

Table A-16

N-Reactor Manual Fire Suppression
Building 181
Fire Area: 30A

AUTOMATIC DETECTION: Rate of rise heat detectors.

AUTOMATIC SUPPRESSION: Individual water spray system - deluge.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility. Fire hydrants approximately 250' from the building.

EQUIPMENT AVAILABLE: Fire extinguishers within the space.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	5.0*	10.0	7.5
2. Alarm	5.0	11.0	8.0
3. Fire Department Site Response	8.5	17.5**	13.0
4. Arrival at the Room of Origin	12.5	26.5	19.0***
5. Finding the Fire	13.0	28.0	20.0
6. Agent Application	13.0	29.0	20.0
7. Extinguishment	14.0	39.0	30.0

NOTES: * Successful control or extinguishment of the fire by means of the deluge water spray system is neglected. Successful operation of the deluge system would limit the fire size at about 1.0 minute after agent discharge (approximately 6.0 minutes).

** At the time of fire department site response, the room is fully involved in fire with room temperature exceeding 1,200°F.

*** At the time that fire department arrives at the room of origin, the room is fully involved in fire with room temperature exceeding 1,200°F.

Table A-17

N-Reactor Manual Fire Suppression
Building 181 Annex, Fire Area: 30B

AUTOMATIC DETECTION: Frangible bulb automatic sprinklers.

AUTOMATIC SUPPRESSION: Deluge system activated by the automatic detection system.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Excellent hydrant accessibility. Fire hydrant approximately 250' from building.

EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0*	13.0**	8.0***
2. Alarm	4.0	13.0	8.0
3. Fire Department Site Response	7.5	19.5	13.0****
4. Arrival at the Room of Origin	11.5	28.5	19.0
5. Finding the Fire	12.0	30.0	20.0
6. Agent Application	12.5*****	32.0	22.0
7. Extinguishment	15.0	40.0	28.0

NOTES: * Successful control or extinguishment of the fire by means of the deluge water spray system is neglected. Successful operation of the deluge system would limit the fire size at about 1.0 minute after agent discharge (approximately 5.0 minutes).

** At the time of detection, the room is fully involved in fire with room temperature exceeding 1,200°F.

*** At the time of detection, the fire size is beyond the capability of fire extinguishers.

**** At the time of fire department site response, the room is fully involved in fire with room temperature exceeding 1,000°F.

***** At the time that fire department starts agent application, the room is fully involved in fire with room temperature exceeding 1,200°F.

Table A-18

N-Reactor Manual Fire Suppression
 Building 182
 Room: El. -24.5, Fire Area: 31A

AUTOMATIC DETECTION: Rate of rise heat detectors for approximately every 900 square feet.

AUTOMATIC SUPPRESSION: Water spray system for diesel engines operated by heat detectors.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility. 1-1/2" hose lines connecting to standpipes.

EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	8.0	60.0	25.0*
2. Alarm	9.0	62.0	26.0
3. Fire Department Site Response	12.5	69.0	31.0
4. Arrival at the Room of Origin	16.5	78.0	37.0
5. Finding the Fire	17.0	80.0	38.0
6. Agent Application	17.0	81.0	38.0
7. Extinguishment	18.0	83.0	39.0

NOTES: * The rate of rise heat detectors are not likely to actuate with given fuel within the space. If they operate, it will be a delayed actuation.

Successful control or extinguishment of the fire by means of the deluge water spray system is neglected. Successful operation of the deluge system would limit the fire size in about 1.0 minute after agent discharge.

Table A-19

N-Reactor Manual Fire Suppression
 Building 182
 Room: El. 0.0, Fire Area: 31B

AUTOMATIC DETECTION: None outside the control room.

AUTOMATIC SUPPRESSION: None outside the control room.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Excellent hydrant accessibility. 1-1/2" standpipe with hose on reel.

EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	10.0*	35.0*	15.0*
2. Alarm	10.5	36.0	16.0
3. Fire Department Site Response	14.0	42.5	21.0
4. Arrival at the Room of Origin	16.0	48.5	25.0
5. Finding the Fire	16.5	49.5	25.5
6. Agent Application	17.0	51.0	26.0
7. Extinguishment	18.0	60.0	30.0

NOTES: * At the time of detection, the fire size is beyond the capability of fire extinguishers.

Table A-20

N-Reactor Manual Fire Suppression
 Building 182
 Room: Control Room @ El. 0.0, Fire Area: 31C

AUTOMATIC DETECTION: Smoke detector.

AUTOMATIC SUPPRESSION: Total flooding Halon system.

HUMAN DETECTION: Excellent/continuously occupied.

MANUAL SUPPRESSION: Excellent hydrant accessibility. 1-1/2" standpipes with hose on reel within the building.

EQUIPMENT AVAILABLE: Fire extinguishers.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	1.0	5.0	2.0*
2. Alarm	1.5	5.5	2.0
3. Fire Department Site Response	5.0	12.0**	7.0
4. Arrival at the Room of Origin	7.0	18.0	11.0
5. Finding the Fire	7.5	19.0	11.5
6. Agent Application	8.0	20.0	12.0**
7. Extinguishment	9.0	22.0	13.0

NOTES: * Successful operation of the total flooding Halon system is neglected. Successful operation of the Halon system would limit the fire size at about 30 seconds after agent discharge (approximately 2.5 minutes).

** At this time, the room is fully involved in fire with room temperature exceeding 1,200°F.

Table A-21

N-Reactor Manual Fire Suppression
 Building 184
 Room: El. 0.0, Fire Area: 32A

AUTOMATIC DETECTION: Automatic sprinklers.

AUTOMATIC SUPPRESSION: Automatic sprinklers.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility. 1-1/2" standpipes within the space with hose.*

EQUIPMENT AVAILABLE: Fire extinguishers within the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	4.0	12.0***	7.0**
2. Alarm	4.5	13.0	8.0
3. Fire Department Site Response	8.0	19.5	13.0
4. Arrival at the Room of Origin	10.0	25.0	17.0
5. Finding the Fire	10.5	27.0	18.0
6. Agent Application	10.5	33.0	22.0
7. Extinguishment	11.0	35.0	23.0

NOTES: * Hose condition is questionable.

** Successful control or extinguishment of the fire by means of automatic sprinklers is neglected. Successful operation of sprinklers would limit the fire size at about 1.0 minute after agent discharge (approximately 8.0 minutes).

*** At the time of detection, the fire size is beyond the capability of fire extinguishers.

Table A-22

N-Reactor Manual Fire Suppression
 Building 184
 Room: El. +22.0, Fire Area: 32A

AUTOMATIC DETECTION: Automatic sprinklers.*

AUTOMATIC SUPPRESSION: Automatic sprinklers.*

HUMAN DETECTION: Excellent. Control room continuously occupied.

MANUAL SUPPRESSION: Good hydrant accessibility. No standpipes.

EQUIPMENT AVAILABLE: Fire extinguishers within the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	2.0	30.0**	10.0**
2. Alarm	2.5	32.0	11.0
3. Fire Department Site Response	6.0	38.0	16.0
4. Arrival at the Room of Origin	8.0	44.0	20.0
5. Finding the Fire	8.5	46.0	21.0
6. Agent Application	8.5	47.0	21.0
7. Extinguishment	9.0	48.0	22.0

NOTES: * Automatic sprinklers may not actuate for a fire involving the existing fuel.

** At the time of detection, the fire size is beyond the capability of fire extinguishers.

Table A-23

N-Reactor Manual Fire Suppression
 Building 184, Annex
 Room: El. O.O, Fire Area: 32B

AUTOMATIC DETECTION: Heat detectors and automatic sprinklers.

AUTOMATIC SUPPRESSION: Automatic sprinkler system and deluge system
 activated by heat detectors.

HUMAN DETECTION: Occasional.

MANUAL SUPPRESSION: Good hydrant accessibility. 1-1/2" hose line on
 reel within the room.

EQUIPMENT AVAILABLE: Fire extinguishers within the room.

EVENT/PHASE DESCRIPTION	CUMULATIVE TIME (MINUTES)		
	Minimum Minutes	Maximum Minutes	Average Minutes
1. Detection	5.0	9.0*	6.5**
2. Alarm	5.5	10.0	6.5
3. Fire Department Site Response	9.0	16.5	11.5
4. Arrival at the Room of Origin	11.0	22.5	15.5
5. Finding the Fire	11.5	24.0	17.0
6. Agent Application	11.5	25.0	17.0
7. Extinguishment	12.5	27.0	18.0

NOTES: * Successful control or extinguishment of the fire by means of
 automatic sprinklers is neglected. Successful operation of
 sprinklers would limit the fire size at about 3.0 minutes
 after agent discharge (approximately 9.5 minutes).

** At the time of detection, the intermediate level heat
 detectors would actuate, triggering the deluge system.
 Extinguishment or control of the fire by means of the deluge
 system is neglected. Successful operation of the deluge
 system would limit the fire size at about 1.0 minute after
 agent discharge.

Figure A-1
 Fire Area 03A - Cable Spreading Room
 Probability of Manual Suppression

Figure A-2

Fire Area 03B - Room 9
Probability of Manual Suppression

Figure A-3

Fire Area 03B - Room 8
Probability of Manual Suppression

Figure A-4

Fire Area 03B - Room 2
Probability of Manual Suppression

Figure A-5

Fire Area 3B1 - Corridor 1
Probability of Manual Suppression

Figure A-6

Fire Area 03G - Room 30
Probability of Manual Suppression

Figure A-7

Fire Area 03J - Corridor 5
Probability of Manual Suppression

Figure A-8
Fire Area 04A - Main Control Room
Probability of Manual Suppression

Figure A-9
Fire Area 21 - Corridor 19
Probability of Manual Suppression

Figure A-10
Fire Area 21 - Building 109 Basement
Probability of Manual Suppression

Figure A-11

Fire Area 21 - Building 109 0 Ft. Turbine Floor
Probability of Manual Suppression

Figure A-12

Fire Areas 26A and 26B - Building 153 - 12 Ft.
Probability of Manual Suppression

Figure A-13

Fire Areas 27A and 27B - Building 153 0 Ft.
Probability of Manual Suppression

Figure A-14
 Fire Area 30A - Building 181
 Probability of Manual Suppression

Figure A-15

Fire Area 30B - Building 181 Annex
Probability of Manual Suppression

Figure A-16
Fire Area 31A - Building 182 Basement
Probability of Manual Suppression

Figure A-17

Fire Area 31B - Building 182.0 Ft.
Probability of Manual Suppression

Figure A-18

Fire Area 31C - Building 182 Control Room
Probability of Manual Suppression

Figure A-19

Fire Area 32A - Building 184 0 Ft.
Probability of Manual Suppression

Figure A-20

Fire Area 32A - Building 184 22 Ft.
Probability of Manual Suppression

Figure A-21

Fire Area 32B - Building 184 Annex
Probability of Manual Suppression

APPENDIX B

N-REACTOR CRITICAL COMPONENTS

BY FIRE AREA

Appendix B

N-Reactor Critical Components by Fire Area

<u>Fire Area</u>	<u>Building</u>	<u>Elevation</u>	<u>Component Description</u>
1D	105N	-15 ft	Emergency Core Cooling System (ECCS) V-4 Outlet Dump Valves and associated cabling in conduit.
3A	105N	-15 Ft	This area contains control cabling for most of the important process and emergency systems for the N-Reactor plant, including the ECCS, High Lift and Fog Spray Diesel pumps, High Pressure Injection pumps, Primary and Secondary Cooling Systems (GSCS), and the Circulating Raw Water Pumps.
3B	105N	-15 Ft	Cabling for both the V-3 ECCS Inlet Valves and the V-4 Outlet Dump Valves is routed through this fire area.
3B1	105N	-15 Ft	Cabling for many of the important plant systems routes through this area on the way to the Cable Spreading and Control Rooms. These include High Lift Diesel and Fog Spray Diesel pumps, High Pressure Injection pumps, PCS pony motors, Dump Condenser Condensate pumps, Low Lift Diesel control, V-4 ECCS valve cabling, GSCS Recirc pumps and associated equipment, and the Circulating Raw Water pumps.
3G	105N	-15 Ft	V-3 valve cabling in cable trays cross this area.
3J	105N	-15 Ft	V-3 valve cabling in cable trays cross this area.
4A	105N	0 Ft	Cabling and controls for all of those systems and components mentioned for Fire Areas 3A and 3B1 above are present in the Main Control Room.
21	109N	-16 Ft	The primary area of concern includes the cable trays in Corridor 19 and the basement area which include cables for the systems and components mentioned for Fire Area 3B1 and the Control Room, Fire Area 4A.

Appendix B

N-Reactor Critical Components by Fire Area (Continued)

Fire Area	Building	Elevation	Component Description
22	109N	-14 Ft	Pressure Transmitters PT-260 and PT-261 associated with the PCS system (PCP suction) are present here.
25A	151N	-14 Ft	Cabling for the systems (CRW pumps) served by the 13.8KV switchgear on the main floor pass through this area.
25B	151N	0 Ft	13.8KV switchgear for CRW pumps 1 and 3 on the "A" bus.
26A	153N	-13 Ft	Conduit and cable trays for the "A" bus systems powered by 4160 volt switchgear.
26B	153N	-13 Ft	Conduit and cable trays for the "B" bus systems powered by 4160 volt switchgear.
27A	153N	0 Ft	"A" bus 4160 volt switchgear for HPI pumps, PCS pony motors, Dump Condenser Condensate pumps, GSCS pumps, and the Instrument and Service Air Compressors.
27B	153N	0 Ft	"B" bus 4160 volt switchgear for same equipment as the "A" side; also includes the "B" bus 13.8KV switchgear for CRW pumps 2 and 4.
30A	181N	421' -0"	Low Diesel Pumps 1 and 2.
30B	181N	421' -0"	Low Lift Diesel Pump 3.
30C	181N	421' -0"	All 4 CRW pumps.
31A	182N	-24' -6"	High Lift Diesel and Fog Spray Diesel pumps, High Pressure Injection pumps, LPFW pumps, and the Master Terminal Box.

Appendix B

N-Reactor Critical Components by Fire Area (Concluded)

<u>Fire Area</u>	<u>Building</u>	<u>Elevation</u>	<u>Component Description</u>
31B	182N	0 Ft	Low Pressure Raw Water pumps, High Pressure Raw Water pumps, Afterheat Removal pumps, High Pressure Filtered Water pumps, Low Pressure Injection pumps, Secondary Loop Makeup pumps.
31C	182N	0 Ft	Bldg. 182 Control Room; MCCS 11 and 12, and controls for equipment in this building.
32A	184N	0 Ft	Instrument and Service Air Compressor.
32B	184N	0 Ft	Auxiliary steam boilers.

APPENDIX C
FIRE EVENT DATA

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
San Onofre 1	2/7/68	Power Operation	Cable	Thermally overloaded 480 V cables caught fire - 55 cables damaged.
San Onofre 1	3/9/68	Power Operation	Cable	Thermally overloaded cables in switchgear room.
Palisades	6/25/71	Cold Shutdown	Air Dryer Filter	Low flow of air through air dryer resulted in temperature buildup and ignition of filter.
LaCrosse	7/15/72	Power Operation	Circulation Pump	Oil on pump lagging ignited ignited by hot pump casing.
Turkey Point 3	12/16/72	Power Operation	Battery Charger	Battery charger overheated and a small fire occurred in the transformer winding insulation.
Robinson 2	4/19/74	Power Operation	Expansion Joint	Cigarette or welding slag from construction workers ignited combustible expansion joint material.
Robinson 2	4/19/74	Power Operation	Expansion Joint	Same type of event as previous event - occurred one week apart.

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires (Continued)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Turkey Point 3	5/75	Power Operation (100%)	Battery Charger	Transformer overheated igniting insulation. Similar to previous event on 12/14/72.
Millstone 2	3/24/76	Hot Shutdown	Motor Control Center	Fire resulted from arcing of a supply lead. Extinguished by de-energizing MCC.
Dresden 2	4/76	Cold Shutdown	Circuit Breaker	ECCS Jockey Pump control feed breaker caught fire from a burned-out contactor coil.
Fitzpatrick	6/11/76	Power Operation (93%)	Circuit Breaker	Overload in HPCI valve circuit breaker. Extinguished by de-energizing breaker.
Millstone 2	11/15/76	Hot Shutdown	Relay-MCC	Relay fire in motor control center.
Pilgrim 1	3/77	Hot Shutdown	Circuit Breaker	Circuit breaker under-voltage coil burnt due to high floating charge on station battery.
Fitzpatrick	4/4/77	Power Operation (88%)	Circuit Breaker	Coil failed by fire in HPCI test valve breaker and extinguished by de-energizing. Similar to 7/28/75 event.

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires (Continued)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Arnold	5/7/77	Refueling Outage	Circuit Breaker	Breaker relay failed, burning open and starting phase burner material above it on fire.
Salem 1	6/30/77	Power Operation	Relay - Cabinet	Fire detection instrumentation panel fire due to relay failure.
Unknown	4/13/78	Power Operation	Circuit Breaker - MCC	Failure breaker contact due to improper maintenance - occurred in motor control center.
Robinson 2	7/16/78	Power Operation	Battery	Resistance heating of terminal connection ignited plastic tops of two cells of a battery.
Unknown	7/27/78	Power Operation	Battery Terminal	Defective terminal or connections not secured.
Arkansas Nuclear One 1	8/16/78	Cold Shutdown	Pump Motor	LPSI pump motor on fire (being used for shutdown cooling) due to incorrect installation of motor bearings resulting in shorting of rotor with the stator.
Salem 1	1/79	Power Operation (95%)	Transformer	Moisture in the windings resulted in a short and subsequent fire.

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires (Continued)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Palisades	4/4/79	Power Operation (100%)	Battery	Battery burst due to internal explosion of hydrogen ignited by a test lead being used to measure voltage.
San Onofre 1	11/27/79	Power Operation (100%)	Switchgear	Rodents shorted two phases of a 480V bus in the switchgear room.
Hatch 2	4/80	Cold Shutdown	Cable	A loose connection resulted in a wire of an RPS motor generator set breaker burning.
Unknown BWR	4/15/80	Power Operation	Bus	Fire involving supply bus occurred in switchgear room.
Peach Bottom 2	6/3/80	Power Operation (100%)	Transformer	A filtering capacitor in a vital bus transformer caught fire damaging the transformer.
Unknown PWR	7/6/80	Power Operation	Circuit Breaker	Circuit breaker caught fire when it failed to close properly because contacts were out of adjustment.
Unknown PWR	10/2/80	Power Operation	Valve Motor	Air sample inlet valve motor issued smoke. Power was removed from motor.

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires (Continued)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Trojan	12/31/80	Power Operation (100%)	Circuit Breaker	Breaker stab misaligned causing ignition of plastic dust collector by arcing.
Palisades	1/24/81	Power Operation (98%)	Pump Motor	Component cooling water pump motor caught fire due to bearing failure from loss of lubricating oil.
San Onofre 1	7/17/81	Cold Shutdown	Gas Decay Tank	Explosion of H ₂ in recombiner.
Indian Point 2	8/10/81	Power Operation (100%)	Pump Motor	Short circuit within SI pump caused fire and an overload trip of its supply breaker.
North Anna 1	11/11/81	Power Operation	Pump	Main feedwater pump fire.
Hatch 1	11/23/81	Cold Shutdown	Relay	Insulation breakdown caused fire in a reactor low-low RPS relay.
Point Beach 1	10/15/82	Power Operation (78%)	Circuit Breaker	Supply breaker for MG set caught fire.

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires (Continued)

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Salem 1	11/9/82	Cold Shutdown	Relay	Relay failure resulted in a fire in a fire detection instrumentation panel. Fire detectors for switchgear rooms, battery room, and DG area were rendered inoperable.
Brunswick 1	11/27/82	Power Operation (68%)	Battery Charger	Resistor on charger amplifier board opened causing a voltage increase and capacitor failure.
Oconee 2	2/3/83	Power Operation (100%)	Pump Motor	Loss of lubrication oil resulted in high bearing temperature and smoke.
Brunswick 1	4/26/83	Refueling	Transformer	Following a loss of offsite power, a fire occurred in a transformer between emergency buses.
Oconee 3	5/25/83	Power Operation (100%)	Cable and Conduit	Welding operation started a fire in conduit surrounding a cable (letdown valve).

Appendix C

N-Reactor Fire Event Data Table - Auxiliary Building Fires (Concluded)

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Salem 2	6/20/83	Cold Shutdown	Transformer	Transformer breaker tripped on overcurrent and was reclosed. Fire occurred immediately thereafter.
Peach Bottom 2	9/9/83	Power Operation (100%)	Control Panel	Water entered a control room ventilation chiller control panel shorting motor starter contactors.
Yankee Rowe	8/2/84	Power Operation (100%)	Circuit Breaker	High resistance in the main disconnecting contacts of the center phase of the breaker caused an arc to propagate to outside phases.

Appendix C

N-Reactor Fire Event Data Table - Reactor Building Fires

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Quad Cities 1	12/10/72	Power Operation	Oil, Insulation	A small open flame was observed within a RHR service-water pump housing. Fire was set by welding sparks on oil soaked insulation.
Peach Bottom 1	12/22/72	Power Operation		The motor on a residual heat removal pump burst into flames due to insufficient lubrication to the lower bearing.
Monticello	5/15/74	Hot Shutdown	Hydrogen	An off-gas ignition occurred resulting in the rupture of both air ejector discharge line rupture discs.
Dresden 3	11/15/74	Power Operation	Hydrogen	An off-gas explosion occurred when the 3A recombiner outlet valve was opened.
Oconee 2	1/31/75	Hot Shutdown	Oil	A small oil fire occurred underneath a reactor coolant pump motor stand.
Brunswick 2	4/14/77	Power Operation	Hydrogen	A hydrogen flame was in the off-gas system at burning the flow orifice or in the jet air ejector.

Appendix C

N-Reactor Fire Event Data Table - Reactor Building Fires (Continued)

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Brunswick 2	6/15/77	Power Operation	Hydrogen	Following an off-gas over-pressurization, a hydrogen fire was detected downstream of the steam jet air ejectors.
Unknown BWR	2/10/78	Power Operation	Electrical	Smoke was noticed coming from a supply breaker.
Indian Point 2	9/4/79	Power Operation	Oil, Insulation	A fire occurred in the reactor coolant pump tube. Insulation was saturated with oil and ignited.
Robinson 2	9/30/79	Power Operation	Oil	Lagging fire on cold leg piping. Fire caused by lubricating oil leak.
San Onofre 1	/16/80	Hot Shutdown	Oil, Insulation	Oil from leaking reactor coolant pump oil filter came in contact with the hot pump casing and ignited.
Nine Mile Point 1	4/22/80	Power Operation	Oil	Fire resulted from lube oil that leaked from a main turbine shaft-driven feed water pump.

Appendix C

N-Reactor Fire Event Data Table - Reactor Building Fires (Concluded)

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Pilgrim	2/24/81	Power Operation	Insulation	A fire was ignited by welded sparks falling on temporary foam rubber insulation.
Unknown PWR	11/7/81	Power Operation	Electrical	Wiring harness was pinched off inside a cabinet and electrically shorted out.
Unknown BWR	2/12/82	Cold Shutdown	Oil	Pipe vibrating loose leaked onto a hot turbine casing.

Appendix C

N-Reactor Fire Event Data Table - Control Room Fires

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Unknown	7/4/78	Power Operation	Diode	Zener diode failed in an RPS circuit.
Three Mile Island 2	7/12/79	Cold Shutdown	Circuit Board	Overheated resistor caused fire in a radiation monitoring readout panel. Extinguished immediately.
Hatch 1*	3/12/83	Power Operation (94%)	Relay	Low reactor water level RPS relay burned causing a 1/2 scram (failed safe). Extinguished by operators.
Hatch 1*	3/30/83	Power Operation (34%)	Relay	Scram discharge volume high level RPS relay burned a 1/2 scram (failed causing safe). Extinguished by operators. Same type of relay as in previous event.

*Counted as one event for quantification of fire frequency.

Appendix C

N-Reactor Fire Event Data Table - Cable Spreading Room Fires

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Browns Ferry 162	3/22/75	Power Operation (100%)	Cable Fire	Spread from cable spreading room to reactor building in Unit 1 and affected Unit 2.
Peach Bottom 3	4/18/77	Power Operation (25%)	Relay Fire	Fire in PCIS logic and RHR valve relay.

Appendix C

N-Reactor Fire Event Data Table - Switchgear Room Fires

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Unknown PWR	11/7/79	Power Operation	480 V Bus	Fire involved 480 V bus; short circuit caused by rodent bridging two energized phases.
Unknown BWR	4/15/80	Power Operation	Bus	Fire involved supply bus in switchgear room.
Unknown PWR	7/6/80	Power Operation	Circuit Breaker	Fire involving switchgear room breaker. Out of adjustment control circuit completed
Yankee Rowe	8/2/84	Power Operation (100%)	Circuit Breaker	A fault occurred in the 480 V supply ACB to bus 4-1; high resistance in the main disconnecting contacts caused an arc to propagate from the center phase to the outside phases.

Appendix C

N-Reactor Fire Event Data Table - Battery Room Fires

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Robinson 2	7/16/78	Power	Battery	Plastic tops of two Operation cells of a station battery caught fire caused by resistance heating of a terminal connection during the heavy dc load of the emergency oil pump.
Unknown	7/27/78	Power Operation	Battery	Fire caused by defective terminal or unsecured connections.
Pallisades	4/4/79	Power Operation (100%)	Battery	A test lead being used to take battery voltage readings fell and struck a battery connector, causing a spark which ignited hydrogen gas.
Brunswick 1	11/27/82	Power Operation (68%)	Capacitor	Battery charger capacitor caught fire for unknown reason.

Appendix C

N-Reactor Fire Event Data Table - Turbine Building Fires

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Nine Mile Point	9/13/72	Power Operation	Oil, Insulation	Leak in oil supply line soaked insulation and ignited when it came in contact with hot pipe.
Yankee Rowe	6/15/73	Power Operation	Oil, Insulation	A fire started in oil soaked insulation around the high pressure turbine bearing casing.
Unknown PWR	8/15/73	Power Operation	Unknown	Fire around turbine area-unknown cause.
Unknown PWR	9/20/74	Power Operation	Ping Pong Balls	Cigarette ignited box of ping pong balls - automatic deluge system initiated.
Kewaunee	4/15/75	Power Operation	Bus	Bus fault resulted in cable insulation damage.
Unknown PWR	6/27/75	Power Operation	Oil	Leaking oil from a turbine oil purifier ignited when it contacted purifier heaters. Cables above the fire charred.
Haddam Neck	9/75	Power	Oil, Operation	Oil soaked insulation Insulation fire on gland steam lines under high pressure turbine.

Appendix C

N-Reactor Fire Event Data Table - Turbine Building Fires (Continued)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Unknown PWR	4/3/77	Power Operation	Hydrogen	Leaking hydrogen at the generator ignited. Purged with CO ₂ by shift personnel.
Saint Lucie 1	4/3/77	Power Operation	Hydrogen	Hydrogen leaked from turbine and ignited. Generator inerted with CO ₂ .
Oyster Creek 1	5/77	Refueling	Cable Insulation	Aluminum to copper bus terminal connectors resulted in high resistance and burned cable insulation.
Peach Bottom 3	9/77	Power Operation	Relays	Three relays in feedwater pump relay cabinet ignited. Since flame retardant cables were used in cabinet, fire did not propagate.
Unknown PWR	7/5/78	Power Operation	Auxiliary Boiler	Class B fire including the auxiliary boiler.
Cook 2	11/13/78	Power Operation	Hydrogen	Hydrogen fire under generator. Purged with CO ₂ .
Browns Ferry 1	1/21/80	Power Operation	Cable Insulation	Fire in cable tray beneath the turbine building operating floor.

Appendix C

N-Reactor Fire Event Data Table - Turbine Building Fires (Concluded)

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Cook 2	12/15/80	Power Operation	Electrical	Fire in generator pilot exciter.
Unknown BWR	7/24/81	Power Operation	Pump	Condensate booster pump binding overheated and caught fire.
Sequoyah 1	1/19/82	Cold Shutdown	Transformer	Neutral ground transformer exploded activating deluge system.
Unknown PWR	2/4/82	Power Operation	Hydrogen	Hydrogen leaked from a bad seal into the generator.
Rancho Seco	3/19/84	Power Operation	Hydrogen	Hydrogen explosion occurred following loss of H ₂ side seal oil pump.
Indian Point 2	10/22/84	Power Operation	Insulation	Fire in insulation at the governor end of the high pressure turbine.
Arnold	11/4/84	Power Operation	Transformer	Transformer fire in yard propagated to the turbine building.

Appendix C

N-Reactor Fire Event Data Table - Diesel Generator Room Fires

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Duane Arnold	3/17/76	Refueling Outage	Oil	The diesel flange gasket leaked exhaust gases with traces of oil onto the exterior of the flange. The oil was ignited by exhaust heat.
Duane Arnold	4/17/76	Power Operation	Oil	Oil leaked onto the diesel exhaust manifold and caught fire.
Millstone 2	9/15/76	Power Operation	Oil, Insulation	A small fire occurred on the exhaust manifold at the control end of the engine.
Zion 2	9/15/76	Power Operation		An operator disconnected a dc tie breaker, tripping the reactor and initiating safe injection. The diesel generator was overloaded resulting in a fire.
Fitzpatrick	10/15/76	Power Operation	Oil	During testing a fire was discovered in the exhaust the emergency diesel generator.
Duane Arnold	11/4/76	Power Operation	Oil	A hairline fracture in a fuel line fitting caused fuel to spray out and be ignited by heat from the exhaust header.

Appendix C

N-Reactor Fire Event Data Table - Diesel Generator Room Fires (Cont'd)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Unknown PWR	12/4/76	Power Operation	Oil	During maintenance on the emergency diesel, a filter was ignited due to overheated oil.
Calvert Cliffs 1	7/11/77	Power Operation	Oil	A small fire developed when lube oil sprayed from the lube oil strainer and ignited on contact with the exhaust manifold.
Kewaunee	9/20/77	Power Operation	Carbon Buildup	A fire was caused by carbon residue buildup in the exhaust path through the turbocharger.
Unknown	12/28/77	Power Operation		Probable cause of fire was combustible materials left in close proximity to the diesel exhaust stack.
Arkansas Nuclear One 1	3/20/78	Refueling Outage	Oil	Failure of bearing oil seal allowed lubricating oil in the turbocharger of the diesel generator.
Arkansas Nuclear One 1	11/15/78	Refueling Outage	Oil	Fire in a diesel exhaust manifold during a test.

Appendix C

N-Reactor Fire Event Data Table - Diesel Generator Room Fires (Cont'd)

<u>Plant Name</u>	<u>Date of Occurrence</u>	<u>Plant Status</u>	<u>Fire Type</u>	<u>Remarks</u>
Crystal River 3	7/24/79	Hot Shutdown	Oil, Carbon	A fire was caused in the exhaust manifold of an emergency diesel generator an excessive fuel-rich mixture aided by oil and carbon accumulation.
Unknown PWR	7/24/79	Power Operation	Electrical	A fire involved the excite control cabinet of a diesel generator.
Crystal River 3	10/15/79	Hot Shutdown	Oil	Fire in the exhaust manifold fuel-oil mix rich on start-(test).
Maine Yankee	10/15/79	Power Operation	Oil	A diesel turbocharger failed which resulted in a fire within the exhaust system.
Calvert Cliffs 2	3/7/80	Power Operation		A small fire occurred in a diesel generator room.
Davis-Besse 1	7/15/80	Power Operation		Fire in a turbocharger.
Davis-Besse 1	9/23/80	Cold Shutdown	Oil	A diesel turbocharger failure which resulted in a fire in the exhaust pipe.

Appendix C

N-Reactor Fire Event Data Table - Diesel Generator Room Fires (Cont'd)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Unknown PWR	3/9/81	Power Operation	Oil, Insulation	Fire involved exhaust manifold insulation.
North Anna 1	4/15/81	Power Operation	Oil	An oil leak in the area of the exhaust manifold started a small fire.
Unknown BWR	5/15/81	Power Operation	Electrical	Smoke filled diesel generator building.
Arkansas Nuclear One 2	6/30/81	Hot Shutdown	Oil, Insulation	Oil leaked through a diesel gasket onto insulation, igniting a fire.
San Onofre 1	7/14/81	Power Operation	Oil	Lube oil spraying from a cracked instrument line was ignited by hot exhaust pipe above the diesel engine.
North Anna 1	7/16/81	Power Operation	Oil	An oil leak in the diesel exhaust manifold caused the fire.
Arkansas Nuclear One 1	7/27/81	Power Operation	Oil, Insulation	Fire on oil-soaked insulation on a diesel engine.

Appendix C

N-Reactor Fire Event Data Table - Diesel Generator Room Fires (Concluded)

Plant Name	Date of Occurrence	Plant Status	Fire Type	Remarks
Zion 1	8/15/81	Power Operation	Oil	Lube oil sprayed passed a operation o-ring seal onto hot exhaust manifold caused fire.
Praire Island 1	8/15/82	Power Operation	Oil	Turbocharger oil gasket filter failure sprayed lube oil onto hot exhaust manifold and ignited.
Peach Bottom 2	9/7/83	Maintenance Outage	Oil	A diesel governor increased fuel flow as a result of a turbo-charger failure. Excess fuel ignited in the exhaust.
Peach Bottom 1	12/18/84	Cold Shutdown	Oil	A diesel fire was caused by a leaking fitting on a fuel injector line.

APPENDIX D

RESPONSE SPECTRA FOR
BUILDINGS 105-N, AND 109-N AND 181-N

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 40'-0"
 el. 51'-0"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-1 N-Reactor 105-N In-Structure Response Acceleration Range 2

Legend:

free-field -22'-0"

foundation -22'-0"

el. 60'-6"

el. 70'-8"

Notes:

Spectra calculated at 5% damping

Accelerations in units of g's

elevation with respect to grade

Figure D-2 N-Reactor 105-N In-Structure Response Acceleration Range 2

Legend:

free-field -22'-0"

foundation -22'-0"

el. 0'-0"

el. 14'-6"

el. 28'-3"

Notes:

Spectra calculated at 5% damping

Accelerations in units of g's

elevation with respect to grade

Figure D-3 N-Reactor 105-N In-Structure Response Acceleration Range 3

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 40'-0"
 el. 51'-0"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-4 N-Reactor 105-N In-Structure Response Acceleration Range 3

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 60'-6"
 el. 70'-8"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-5 N-Reactor 105-N In-Structure Response Acceleration Range 3

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. -16'-0"
 el. 0'-0"
 el. 26'-0"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-6 N-Reactor 109-N In-Structure Response Acceleration Range 1

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 39'-6"
 el. 60'-6"
 el. 87'-6"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-7 N-Reactor 109-N In-Structure Response Acceleration Range 1

Legend:

free-field -22'-0"	_____
foundation -22'-0"	_____
el. -16'-0"	_____
el. 0'-0"	_____
el. 26'-0"	_____

Notes:
 Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-8 N-Reactor 109-N In-Structure Response Acceleration Range 2

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 39'-6"
 el. 60'-6"
 el. 87'-6"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-9 N-Reactor 109-N In-Structure Response Acceleration Range 2

Legend:

free-field -22'-0"

foundation -22'-0"

el. -16'-0"

el. 0'-0"

el. 26'-0"

Notes:

Spectra calculated at 5% damping
Accelerations in units of g's
elevation with respect to grade

Figure D-10 N-Reactor 109-N In-Structure Response Acceleration Range 3

Legend:

free-field -22'-0"
 foundation -22'-0"
 el. 39'-6"
 el. 60'-6"
 el. 87'-6"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's
 elevation with respect to grade

Figure D-11 N-Reactor 109-N In-Structure Response Acceleration Range 3

Legend:

free-field 454'-0"

foundation 355'-0"

el. 390'-0"

el. 421'-0"

el. 439'-0"

Notes:

Spectra calculated at 5% damping

Accelerations in units of g's

Figure D-12 N-Reactor 181-N In-Structure Response Acceleration Range 2

Legend:

free-field 454'-0"

foundation 355'-0"

Gantry Crane

Notes:

Spectra calculated at 5% damping
Accelerations in units of g's

Figure D-13 N-Reactor 181-N In-Structure Response Acceleration Range 2

Legend:

free-field 454'-0"
 foundation 355'-0"
 el. 390'-0"
 el. 421'-0"
 el. 439'-0"

Notes:

Spectra calculated at 5% damping
 Accelerations in units of g's

Figure D-14 N-Reactor 181-N In-Structure Response Acceleration Range 3

Legend:

free-field 454'-0"

foundation 355'-0"

Gantry Crane

Notes:

Spectra calculated at 5% damping

Accelerations in units of g's

Figure D-15 N-Reactor 181-N In-Structure Response Acceleration Range 3

Table D-1

Free-Field Median Response
Acceleration Range 1 (g)

<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
E-W	0.57	0.43	0.44	0.34	0.25
N-S	0.55	0.45	0.46	0.37	0.25
Ver	0.34	0.33	0.33	0.29	0.17

Table D-2

Free-Field Median Response
Acceleration Range 2 (g)

<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
E-W	1.15	0.87	0.89	0.69	0.50
N-S	1.11	0.89	0.92	0.74	0.50
Ver	0.68	0.66	0.66	0.58	0.33

Table D-3

Free-Field Median Response
Acceleration Range 3 (g)

<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
E-W	1.72	1.30	1.33	1.03	0.75
N-S	1.66	1.34	1.38	1.11	0.75
Ver	1.01	0.98	0.99	0.87	0.50

Table D-4

**Building 105-N In-Structure Response
Acceleration Range 1 (g)**

<u>Fl.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	0.67	0.38	0.39	0.32	0.27
-22'-0"	N-S	0.66	0.38	0.38	0.32	0.27
-22'-0"	Ver	0.44	0.38	0.40	0.29	0.20
0'-0"	E-W	0.69	0.39	0.39	0.32	0.28
0'-0"	N-S	0.70	0.40	0.41	0.34	0.29
0'-0"	Ver	0.48	0.40	0.41	0.30	0.21
14'-6"	E-W	0.71	0.41	0.41	0.34	0.29
14'-6"	N-S	0.74	0.43	0.43	0.36	0.30
14'-6"	Ver	0.48	0.40	0.42	0.31	0.21
28'-3"	E-W	0.74	0.42	0.43	0.35	0.30
28'-3"	N-S	0.76	0.45	0.45	0.38	0.31
28'-3"	Ver	0.48	0.41	0.42	0.31	0.21
40'-0"	E-W	0.75	0.43	0.44	0.36	0.30
40'-0"	N-S	0.78	0.46	0.46	0.39	0.31
40'-0"	Ver	0.48	0.41	0.43	0.32	0.21
51'-0"	E-W	0.77	0.45	0.45	0.38	0.31
51'-0"	N-S	0.81	0.48	0.48	0.41	0.32
51'-0"	Ver	0.48	0.41	0.43	0.32	0.21
60'-6"	E-W	0.78	0.46	0.46	0.39	0.31
60'-6"	N-S	0.82	0.49	0.49	0.42	0.33
60'-6"	Ver	0.48	0.42	0.43	0.32	0.22
70'-8"	E-W	0.79	0.46	0.47	0.39	0.32
70'-8"	N-S	0.84	0.50	0.50	0.43	0.33
70'-8"	Ver	0.49	0.42	0.43	0.33	0.22

Table D-5

Building 105-N Median In-Structure Response
Acceleration Range 2 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	1.15	0.73	0.74	0.61	0.52
-22'-0"	N-S	1.13	0.70	0.72	0.59	0.51
-22'-0"	Ver	0.86	0.63	0.64	0.50	0.37
0'-0"	E-W	1.17	0.74	0.75	0.62	0.54
0'-0"	N-S	1.18	0.73	0.75	0.62	0.53
0'-0"	Ver	0.94	0.67	0.68	0.53	0.38
14'-6"	E-W	1.20	0.76	0.78	0.64	0.55
14'-6"	N-S	1.21	0.77	0.78	0.65	0.54
14'-6"	Ver	0.94	0.68	0.69	0.54	0.39
28'-3"	E-W	1.23	0.79	0.80	0.66	0.56
28'-3"	N-S	1.25	0.79	0.80	0.67	0.56
28'-3"	Ver	0.93	0.68	0.69	0.55	0.39
40'-0"	E-W	1.25	0.80	0.81	0.67	0.57
40'-0"	N-S	1.27	0.80	0.81	0.69	0.57
40'-0"	Ver	0.94	0.69	0.70	0.55	0.39
51'-0"	E-W	1.28	0.82	0.83	0.70	0.58
51'-0"	N-S	1.30	0.82	0.83	0.71	0.58
51'-0"	Ver	0.93	0.69	0.70	0.55	0.39
60'-6"	E-W	1.30	0.83	0.85	0.71	0.59
60'-6"	N-S	1.32	0.83	0.84	0.72	0.59
60'-6"	Ver	0.94	0.69	0.71	0.56	0.39
70'-8"	E-W	1.31	0.84	0.85	0.71	0.59
70'-8"	N-S	1.34	0.84	0.85	0.74	0.60
70'-8"	Ver	0.95	0.70	0.71	0.57	0.40

Table D-6

Building 105-N Median In-Structure Response
Acceleration Range 3 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	1.44	0.96	0.98	0.80	0.73
-22'-0"	N-S	1.42	0.95	0.97	0.80	0.73
-22'-0"	Ver	1.21	0.79	0.80	0.66	0.47
0'-0"	E-W	1.44	0.96	0.98	0.81	0.74
0'-0"	N-S	1.45	0.97	1.00	0.82	0.74
0'-0"	Ver	1.30	0.86	0.87	0.71	0.51
14'-6"	E-W	1.47	0.99	1.00	0.83	0.76
14'-6"	N-S	1.47	1.00	1.02	0.84	0.75
14'-6"	Ver	1.30	0.87	0.88	0.72	0.51
28'-3"	E-W	1.49	1.01	1.02	0.85	0.77
28'-3"	N-S	1.49	1.01	1.04	0.86	0.77
28'-3"	Ver	1.30	0.87	0.88	0.73	0.51
40'-0"	E-W	1.51	1.02	1.03	0.86	0.78
40'-0"	N-S	1.50	1.02	1.04	0.87	0.77
40'-0"	Ver	1.31	0.89	0.89	0.74	0.52
51'-0"	E-W	1.53	1.04	1.05	0.88	0.79
51'-0"	N-S	1.52	1.03	1.05	0.88	0.78
51'-0"	Ver	1.30	0.88	0.89	0.74	0.52
60'-6"	E-W	1.54	1.05	1.06	0.89	0.80
60'-6"	N-S	1.53	1.03	1.05	0.89	0.78
60'-6"	Ver	1.31	0.89	0.90	0.75	0.52
70'-8"	E-W	1.55	1.05	1.06	0.90	0.80
70'-8"	N-S	1.54	1.04	1.05	0.90	0.79
70'-8"	Ver	1.32	0.90	0.91	0.75	0.53

Table D-7

Building 109-N In-Structure Response
Acceleration Range 1 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	0.67	0.38	0.39	0.32	0.27
-22'-0"	N-S	0.66	0.38	0.38	0.32	0.27
-22'-0"	Ver	0.44	0.38	0.40	0.29	0.20
-16'-0"	E-W	0.68	0.38	0.39	0.31	0.27
-16'-0"	N-S	0.66	0.38	0.38	0.32	0.27
-16'-0"	Ver	0.41	0.40	0.42	0.29	0.20
0'-0"	E-W	0.70	0.39	0.40	0.32	0.28
0'-0"	N-S	0.70	0.40	0.41	0.34	0.29
0'-0"	Ver	0.42	0.41	0.43	0.30	0.20
26'-0"	E-W	0.74	0.43	0.43	0.35	0.29
26'-0"	N-S	0.76	0.44	0.45	0.38	0.30
26'-0"	Ver	0.42	0.42	0.44	0.31	0.20
39'-6"	E-W	0.75	0.44	0.44	0.36	0.30
39'-6"	N-S	0.78	0.46	0.46	0.39	0.31
39'-6"	Ver	0.42	0.42	0.44	0.31	0.20
60'-6"	E-W	0.78	0.46	0.47	0.39	0.31
60'-6"	N-S	0.82	0.49	0.49	0.42	0.33
60'-6"	Ver	0.44	0.44	0.45	0.34	0.22
87'-6"	E-W	0.82	0.51	0.51	0.45	0.33
87'-6"	N-S	0.88	0.54	0.54	0.49	0.36
87'-6"	Ver	0.45	0.45	0.46	0.36	0.22

Table D-8

Building 109-N Median In-Structure Response
Acceleration Range 2 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	1.15	0.73	0.74	0.61	0.52
-22'-0"	N-S	1.13	0.70	0.72	0.59	0.51
-22'-0"	Ver	0.86	0.63	0.64	0.50	0.37
-16'-0"	E-W	1.17	0.73	0.74	0.61	0.52
-16'-0"	N-S	1.13	0.69	0.71	0.59	0.51
-16'-0"	Ver	0.84	0.66	0.68	0.51	0.37
0'-0"	E-W	1.19	0.74	0.76	0.62	0.53
0'-0"	N-S	1.17	0.73	0.74	0.62	0.53
0'-0"	Ver	0.85	0.67	0.69	0.51	0.37
26'-0"	E-W	1.25	0.79	0.81	0.66	0.55
26'-0"	N-S	1.24	0.78	0.80	0.67	0.56
26'-0"	Ver	0.85	0.68	0.70	0.52	0.37
39'-6"	E-W	1.27	0.81	0.82	0.67	0.56
39'-6"	N-S	1.27	0.80	0.81	0.69	0.57
39'-6"	Ver	0.85	0.68	0.70	0.52	0.37
60'-6"	E-W	1.31	0.84	0.85	0.71	0.58
60'-6"	N-S	1.32	0.83	0.84	0.72	0.59
60'-6"	Ver	0.87	0.71	0.72	0.57	0.39
87'-6"	E-W	1.36	0.90	0.91	0.79	0.61
87'-6"	N-S	1.39	0.90	0.90	0.82	0.62
87'-6"	Ver	0.88	0.72	0.73	0.59	0.40

Table D-9

Building 109-N Median In-Structure Response
Acceleration Range 3 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
-22'-0"	E-W	1.44	0.96	0.98	0.80	0.73
-22'-0"	N-S	1.42	0.95	0.97	0.80	0.73
-22'-0"	Ver	1.21	0.79	0.80	0.66	0.47
-16'-0"	E-W	1.48	0.96	0.98	0.80	0.73
-16'-0"	N-S	1.42	0.94	0.96	0.79	0.73
-16'-0"	Ver	1.21	0.81	0.82	0.67	0.49
0'-0"	E-W	1.50	0.98	0.99	0.82	0.74
0'-0"	N-S	1.44	0.97	0.99	0.82	0.74
0'-0"	Ver	1.22	0.82	0.83	0.68	0.49
26'-0"	E-W	1.55	1.02	1.04	0.85	0.77
26'-0"	N-S	1.48	1.01	1.03	0.85	0.76
26'-0"	Ver	1.23	0.83	0.84	0.69	0.49
39'-6"	E-W	1.56	1.03	1.05	0.86	0.78
39'-6"	N-S	1.50	1.02	1.04	0.87	0.77
39'-6"	Ver	1.22	0.83	0.84	0.69	0.49
60'-6"	E-W	1.58	1.05	1.07	0.89	0.79
60'-6"	N-S	1.53	1.03	1.05	0.89	0.78
60'-6"	Ver	1.24	0.86	0.86	0.74	0.50
87'-6"	E-W	1.63	1.11	1.13	0.97	0.82
87'-6"	N-S	1.58	1.08	1.09	0.98	0.80
87'-6"	Ver	1.25	0.88	0.87	0.76	0.50

Table D-10
Building 181-N Median In-Structure Response
Acceleration Range 1 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
355'-0"	E-W	0.35	0.32	0.34	0.24	0.18
355'-0"	N-S	0.38	0.31	0.32	0.26	0.19
355'-0"	Ver	0.36	0.35	0.36	0.30	0.18
390'-0"	E-W	0.38	0.44	0.46	0.35	0.20
390'-0"	N-S	0.58	0.48	0.50	0.33	0.27
390'-0"	Ver	0.36	0.37	0.37	0.32	0.20
421'-6"	E-W	0.40	0.53	0.54	0.44	0.23
421'-6"	N-S	0.67	0.69	0.73	0.48	0.32
421'-6"	Ver	0.37	0.37	0.38	0.33	0.21
439'-0"	E-W	0.46	1.00	1.01	0.98	0.37
439'-0"	N-S	0.98	1.56	1.69	0.96	0.59
439'-0"	Ver	0.37	0.39	0.39	0.36	0.25
Crane	E-W	0.92	0.59	0.60	0.45	0.38
Crane	N-S	1.02	0.56	0.56	0.49	0.45

Table D-11

Building 181-N Median In-Structure Response
Acceleration Range 2 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
355'-0"	E-W	0.72	0.64	0.67	0.49	0.36
355'-0"	N-S	0.76	0.61	0.63	0.51	0.36
355'-0"	Ver	0.73	0.69	0.70	0.60	0.36
390'-0"	E-W	0.80	0.88	0.91	0.66	0.40
390'-0"	N-S	1.13	0.88	0.91	0.64	0.50
390'-0"	Ver	0.74	0.72	0.73	0.63	0.37
421'-6"	E-W	0.85	1.09	1.11	0.86	0.45
421'-6"	N-S	1.31	1.25	1.31	0.89	0.60
421'-6"	Ver	0.74	0.73	0.74	0.65	0.39
439'-0"	E-W	0.99	1.99	2.01	1.85	0.73
439'-0"	N-S	1.87	2.66	2.84	1.65	1.06
439'-0"	Ver	0.75	0.77	0.77	0.71	0.43
Crane	E-W	1.88	1.20	1.22	0.93	0.78
Crane	N-S	1.91	1.03	1.04	0.90	0.82

Table D-12

Building 181-N Median In-Structure Response
Acceleration Range 3 (g)

<u>El.</u>	<u>Dir.</u>	<u>2-5 (Hz)</u>	<u>5-10 (Hz)</u>	<u>7 (Hz)</u>	<u>10 (Hz)</u>	<u>zpa</u>
355'-0"	E-W	1.17	0.96	0.98	0.74	0.53
355'-0"	N-S	1.16	0.94	0.95	0.80	0.54
355'-0"	Ver	1.10	1.01	1.02	0.86	0.52
390'-0"	E-W	1.33	1.28	1.33	0.91	0.61
390'-0"	N-S	1.79	1.30	1.34	0.95	0.73
390'-0"	Ver	1.11	1.04	1.05	0.90	0.54
421'-6"	E-W	1.45	1.63	1.69	1.20	0.68
421'-6"	N-S	2.12	1.85	1.90	1.37	0.91
421'-6"	Ver	1.12	1.06	1.07	0.93	0.56
439'-0"	E-W	1.70	2.99	3.08	2.44	1.07
439'-0"	N-S	3.09	3.83	4.03	2.52	1.64
439'-0"	Ver	1.14	1.13	1.13	1.02	0.60
Crane	E-W	3.13	1.94	1.98	1.48	1.23
Crane	N-S	2.78	1.47	1.47	1.26	1.16

APPENDIX E

N-REACTOR COMPUTATIONAL INPUT PARAMETERS

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N-REACTOR FRAGILITY FILE (2/1/90)

Category	Median	Beta-r	Beta-u	Component
1	0.25	0.25	.25	CERAMIC INSULATORS
2	4.00	0.48	.75	RELAY CHATTER
3	7.63	0.48	.74	CIRCUIT BREAKER TRIP
4	2.50	0.40	.39	BATTERIES
5	2.29	0.31	.39	BATTERY RACKS
6	2.00	0.26	.35	INVERTORS
7	8.80	0.28	.30	TRANSFORMERS
8	7.63	0.48	.74	MOTOR CONTROL CENTER
9	7.63	0.48	.66	AUX RELAY CABINET
10	6.43	0.29	.66	SWITCHGEAR
11	2.23	0.34	.19	CABLE TRAYS
12	11.50	0.46	.74	CONTROL PANELS AND RACKS
13	7.68	0.20	.35	LOCAL INSTRUMENTS
14	1.00	0.25	.31	DIESEL GENERATOR
15	12.10	0.27	.31	MOTORS-HORIZONTAL
16	2.80	0.25	.27	MOTOR-DRIVEN PUMPS & COMPRESSORS
17	2.21	0.22	.32	LG. VERT. M-D. CENTRIF PUMP
18	6.50	0.26	.60	LMOV
19	4.83	0.26	.60	SMALL MOV & AOVs
20	6.50	0.26	.34	LG. PNEUM/HYD VALVE
21	8.90	0.20	.35	LG. MANUAL,CHECK,RELIEF VALVE
22	12.50	0.33	.43	MISC. SMALL VALVES
23	3.00	0.30	.53	LG. HORIZ. VESSELS
24	1.84	0.25	.45	SM-MED HEAT EXCHANGERS & VESSELS
25	1.46	0.20	.35	LG. VERT VESSELS w/ FORMED HEADS
26	0.45	0.35	.29	LG. VERT. FLAT BOTTOMED TANKS
27	6.90	0.62	.25	**fan coolers
28	5.10	0.23	.25	**turbines (hpci & rcic)
29	2.45	0.24	.30	**steam generator (zion ssmrp)
30	3.21	0.51	.30	**small vertical tanks
31	7.68	0.26	.60	**small solenoid valves
32	99.00	0.30	.30	**dummy - no seismic failure
33	2.00	0.35	.30	**trash racks - 181 intake
34	1.00	0.03	.23	**BUILDING 105
35	1.70	0.03	.23	**BUILDING 109
36	0.90	0.03	.30	**PRIMARY SHIELD WALL
37	0.47	0.08	.20	**BUILDING 181-N
38	0.54	0.06	.21	**CONCRETE SILO
39	1.25	0.06	.30	**DEMINERALIZED WATER TANK
40	5.00	0.33	.35	**annunciators
41	7.68	0.20	.35	**switches (local instruments)
42	0.01	0.30	.30	**dummy - causes failure
43	0.95892	0.50	.30	SLOCA FIT-SSMRP
44	1.025	0.147	.273	**BUILDING 182-N
45	99.00	0.40	.30	LLOCA FIT-SSMRP
46	0.76	0.25	.30	13.8 KV BUS
47	0.76	0.25	.30	4 KV BUS
48	0.35	0.147	.299	**BUILDING 109NT
49	0.72	0.18	.33	**CORE INLET PIPING
50	0.25	0.06	.3	**BLDG 151
51	0.25	0.06	.3	**BLDG 153
52	0.25	0.06	.3	**BLDG 184

N REACTOR RESPONSE FILE (2/1/90)

Category	Median	Beta-r	Beta-u	Response Location
1	1.00	0.25	.25	FREE-FIELD ZPA
2	2.32	0.45	.25	2-5 HZ
3	1.80	0.35	.25	5-10
4	2.60	0.35	.25	3
5	1.12	0.35	.25	105-N (-) 22'-0" ZPA
6	1.72	0.45	.25	5-10 HZ
7	1.16	0.35	.25	0'-0" ZPA
8	1.72	0.45	.25	5-10 HZ
9	1.20	0.35	.25	14'-6" ZPA
10	1.24	0.35	.25	28'-3" ZPA
11	1.24	0.35	.25	40'-0" ZPA
12	1.28	0.35	.25	51'-0" ZPA
13	1.32	0.35	.25	60'-6" ZPA
14	1.08	0.35	.25	109-N (-) 16'-0" ZPA
15	3.16	0.45	.25	2-5 HZ
16	3.60	0.45	.25	3 HZ
17	1.72	0.45	.25	5-10 HZ
18	1.12	0.35	.25	0'-0" ZPA
19	2.92	0.45	.25	2-5 HZ
20	3.66	0.45	.25	3 HZ
21	1.76	0.45	.25	5-10 HZ
22	1.20	0.35	.25	26'-0" ZPA
23	0.88	0.35	.25	109-NT (-) 16'-0" ZPA
24	2.12	0.45	.25	5-10 HZ
25	1.24	0.35	.25	0'-0" ZPA
26	2.12	0.45	.25	5-10 HZ
27	2.15	0.35	.25	15'-3" ZPA
28	1.88	0.35	.25	36'-8" ZPA
29	0.80	0.35	.25	151-N& (-) 16'-0", (-) 24'-0" ZPA
30	1.64	0.45	.25	153-N& 5-10 HZ
31	0.84	0.35	.25	182-N 0'-0" ZPA
32	2.08	0.45	.25	5-10 HZ
33	1.96	0.35	.25	16'-0", 24'-0" ZPA
34	0.72	0.35	.25	181-N El. 360'-0" ZPA
35	1.50	0.45	.25	3-7 HZ
36	1.28	0.45	.25	5-10 HZ
37	1.08	0.35	.25	El. 390'-0" ZPA
38	2.48	0.45	.25	3-7 HZ
39	2.02	0.45	.25	5-10 HZ
40	1.28	0.35	.25	El. 421'-0" ZPA
41	3.04	0.45	.25	[= 0'] 3-7 HZ
42	2.88	0.45	.25	5-10 HZ
43	2.40	0.35	.25	El. 440'-0" ZPA
44	6.68	0.45	.25	5-10 HZ
45	8.16	0.45	.25	6 HZ N-S COMP
46	1.24	0.35	.25	184-N 0'-0" ZPA
47	2.12	0.45	.25	5-10 HZ
48	1.24	0.35	.25	109-N 39'-6" ZPA
49	1.28	0.35	.25	60'-6" ZPA
50	1.36	0.35	.25	184-N 15'-3" ZPA
51	1.88	0.35	.25	36'-8" ZPA
52	2.80	0.45	.25	15'-3" 5-10 HZ
53	1.56	0.45	.25	105-N (-) 16'-0" 5-10 HZ (ROOM 6)

N-REACTOR CROSS REFERENCE FILE FOR SEISMIC ANALYSIS (2/1/90)

Random Probability	Basic Event Name	Error Factor	Frag Cat	Resp Cat	Corr Coeff	Event Number
8.617E-11	\$ A002MOD	\$ 3.0	32	1		1
1.200E-05	\$ AAG19--D	\$	21	5		2
2.500E-02	\$ AAG3-1-A	\$	20	5		3
7.920E-04	\$ AAG3-1-C	\$	20	5		4
2.500E-02	\$ AAG3-2-A	\$	20	5		5
7.920E-04	\$ AAG3-2-C	\$	20	5		6
2.500E-02	\$ AAG3-3-A	\$	20	5		7
7.920E-04	\$ AAG3-3-C	\$	20	5		8
2.500E-02	\$ AAG3-4-A	\$	20	5		9
7.920E-04	\$ AAG3-4-C	\$	20	5		10
2.500E-02	\$ AAG3-5-A	\$	20	5		11
7.920E-04	\$ AAG3-5-C	\$	20	5		12
2.500E-02	\$ AAG3-6-A	\$	20	5		13
7.920E-04	\$ AAG3-6-C	\$	20	5		14
2.500E-02	\$ AAG3-7-A	\$	20	5		15
7.920E-04	\$ AAG3-7-C	\$	20	5		16
2.500E-02	\$ AAG3-8-A	\$	20	5		17
7.920E-04	\$ AAG3-8-C	\$	20	5		18
2.500E-02	\$ AAG4-1-A	\$	20	7		19
7.920E-04	\$ AAG4-1-C	\$	20	7		20
0.000E-01	\$ AAG4-2-A	\$	20	7		21
7.920E-04	\$ AAG4-2-C	\$	20	7		22
1.000E+00	\$ AAG4-2-X	\$	32	7		23
2.500E-02	\$ AAG4-3-A	\$	20	7		24
7.920E-04	\$ AAG4-3-C	\$	20	7		25
2.500E-02	\$ AAG4-4-A	\$	20	7		26
7.920E-04	\$ AAG4-4-C	\$	20	7		27
2.500E-02	\$ AAG4-5-A	\$	20	7		28
7.920E-04	\$ AAG4-5-C	\$	20	7		29
2.500E-02	\$ AAG4-6-A	\$	20	7		30
7.920E-04	\$ AAG4-6-C	\$	20	7		31
2.500E-02	\$ AAG4-7-A	\$	20	7		32
7.920E-04	\$ AAG4-7-C	\$	20	7		33
2.500E-02	\$ AAG4-8-A	\$	20	7		34
7.920E-04	\$ AAG4-8-C	\$	20	7		35
5.600E-04	\$ AAGV3CMA	\$	20	7		36
5.600E-04	\$ AAGV4CMA	\$	20	7		37
1.248E-04	\$ AARHL1-L	\$	24	29		38
1.248E-04	\$ AARHL2-L	\$	24	29		39
1.248E-04	\$ AARHL3-L	\$	24	29		40
1.248E-04	\$ AARLL1-L	\$	24	43		41
1.248E-04	\$ AARLL2-L	\$	24	43		42
1.248E-04	\$ AARLL3-L	\$	24	43		43
2.700E-05	\$ ACV1-1-A	\$	21	7		44
2.700E-05	\$ ACV1-2-A	\$	21	7		45
2.700E-05	\$ ACV1-3-A	\$	21	7		46
2.700E-05	\$ ACV1-4-A	\$	21	7		47
2.700E-05	\$ ACV1-5-A	\$	21	7		48
2.700E-05	\$ ACV1-6-A	\$	21	7		49
2.700E-05	\$ ACV1-7-A	\$	21	7		50
2.700E-05	\$ ACV1-8-A	\$	21	7		51
1.320E-04	\$ ACV5-1-L	\$	21	12		52
1.320E-04	\$ ACV5-2-L	\$	21	12		53
1.320E-04	\$ ACV5-3-L	\$	21	12		54
1.320E-04	\$ ACV5-4-L	\$	21	12		55
1.320E-04	\$ ACV5-5-L	\$	21	12		56
1.320E-04	\$ ACV5-6-L	\$	21	12		57
1.320E-04	\$ ACV5-7-L	\$	21	12		58
1.320E-04	\$ ACV5-8-L	\$	21	12		59

1.320E-04	\$	ACV5-9-L	\$	21	12	60
2.700E-05	\$	ACV5001A	\$	22	40	61
2.700E-05	\$	ACV5002A	\$	22	40	62
2.700E-05	\$	ACV5035A	\$	22	40	63
2.700E-05	\$	ACV5036A	\$	22	40	64
2.700E-05	\$	ACV504-A	\$	22	43	65
2.700E-05	\$	ACV5041A	\$	21	37	66
2.700E-05	\$	ACV5042A	\$	21	37	67
2.700E-05	\$	ACV510-A	\$	21	40	68
1.320E-04	\$	ACV510-L	\$	21	11	69
1.320E-04	\$	ACV511-L	\$	21	11	70
1.320E-04	\$	ACV512-L	\$	21	11	71
1.320E-04	\$	ACV513-L	\$	21	11	72
1.320E-04	\$	ACV514-L	\$	21	11	73
1.320E-04	\$	ACV515-L	\$	21	11	74
1.320E-04	\$	ACV516-L	\$	21	11	75
2.700E-05	\$	ACV517-A	\$	22	40	76
2.700E-05	\$	ACV518-A	\$	22	40	77
2.700E-05	\$	ACV5191A	\$	21	40	78
2.700E-05	\$	ACV5192A	\$	21	40	79
2.700E-05	\$	ACV5543A	\$	22	31	80
2.700E-05	\$	ACV5544A	\$	22	31	81
2.700E-05	\$	ACV5545A	\$	22	31	82
2.700E-05	\$	ACV5621A	\$	21	29	83
2.700E-05	\$	ACV5622A	\$	21	29	84
2.700E-05	\$	ACV5623A	\$	21	29	85
2.700E-05	\$	ACV5731A	\$	22	31	86
2.700E-05	\$	ACV5732A	\$	22	31	87
2.700E-05	\$	ACV5733A	\$	22	31	88
2.700E-05	\$	ACV601-A	\$	21	5	89
2.700E-05	\$	ACV602-A	\$	21	5	90
2.700E-05	\$	ACV6181A	\$	21	31	91
2.700E-05	\$	ACV6182A	\$	21	31	92
2.700E-05	\$	ACV6183A	\$	21	31	93
2.700E-05	\$	ACV812-A	\$	21	1	94
2.700E-05	\$	ACV820-A	\$	21	1	95
2.700E-05	\$	ACVA31-A	\$	22	5	96
2.700E-05	\$	ACVA32-A	\$	22	5	97
2.700E-05	\$	ACVA33-A	\$	22	5	98
2.700E-05	\$	ACVA34-A	\$	22	5	99
2.700E-05	\$	ACVA35-A	\$	22	5	100
2.700E-05	\$	ACVA36-A	\$	22	5	101
2.700E-05	\$	ACVA37-A	\$	22	5	102
2.700E-05	\$	ACVA38-A	\$	22	5	103
2.700E-05	\$	ACVA41-A	\$	22	7	104
2.700E-05	\$	ACVA42-A	\$	22	7	105
2.700E-05	\$	ACVA43-A	\$	22	7	106
2.700E-05	\$	ACVA44-A	\$	22	7	107
2.700E-05	\$	ACVA45-A	\$	22	7	108
2.700E-05	\$	ACVA46-A	\$	22	7	109
2.700E-05	\$	ACVA47-A	\$	22	7	110
2.700E-05	\$	ACVA48-A	\$	22	7	111
2.700E-05	\$	ACVB31-A	\$	22	5	112
2.700E-05	\$	ACVB32-A	\$	22	5	113
2.700E-05	\$	ACVB33-A	\$	22	5	114
2.700E-05	\$	ACVB34-A	\$	22	5	115
2.700E-05	\$	ACVB35-A	\$	22	5	116
2.700E-05	\$	ACVB36-A	\$	22	5	117
2.700E-05	\$	ACVB37-A	\$	22	5	118
2.700E-05	\$	ACVB38-A	\$	22	5	119

2.700E-05	\$	ACVB41-A	\$	22	7	120
2.700E-05	\$	ACVB42-A	\$	22	7	121
2.700E-05	\$	ACVB43-A	\$	22	7	122
2.700E-05	\$	ACVB44-A	\$	22	7	123
2.700E-05	\$	ACVB45-A	\$	22	7	124
2.700E-05	\$	ACVB46-A	\$	22	7	125
2.700E-05	\$	ACVB47-A	\$	22	7	126
2.700E-05	\$	ACVB48-A	\$	22	7	127
2.700E-05	\$	ACVC31-A	\$	22	5	128
2.700E-05	\$	ACVC32-A	\$	22	5	129
2.700E-05	\$	ACVC33-A	\$	22	5	130
2.700E-05	\$	ACVC34-A	\$	22	5	131
2.700E-05	\$	ACVC35-A	\$	22	5	132
2.700E-05	\$	ACVC36-A	\$	22	5	133
2.700E-05	\$	ACVC37-A	\$	22	5	134
2.700E-05	\$	ACVC38-A	\$	22	5	135
2.700E-05	\$	ACVC41-A	\$	22	7	136
2.700E-05	\$	ACVC42-A	\$	22	7	137
2.700E-05	\$	ACVC43-A	\$	22	7	138
2.700E-05	\$	ACVC44-A	\$	22	7	139
2.700E-05	\$	ACVC45-A	\$	22	7	140
2.700E-05	\$	ACVC46-A	\$	22	7	141
2.700E-05	\$	ACVC47-A	\$	22	7	142
2.700E-05	\$	ACVC48-A	\$	22	7	143
2.700E-05	\$	ACVR518A	\$	21	40	144
1.600E-04	\$	ADPHL1-F	\$	16	29	145
7.600E-02	\$	ADPHL1-G	\$	32	29	146
1.600E-04	\$	ADPHL2-F	\$	16	29	147
7.600E-02	\$	ADPHL2-G	\$	32	29	148
1.600E-04	\$	ADPHL3-F	\$	16	29	149
7.600E-02	\$	ADPHL3-G	\$	32	29	150
1.700E-04	\$	ADPLL1-F	\$	17	40	151
6.700E-03	\$	ADPLL1-G	\$	17	40	152
1.700E-04	\$	ADPLL2-F	\$	17	40	153
6.700E-03	\$	ADPLL2-G	\$	17	40	154
1.700E-04	\$	ADPLL3-F	\$	17	40	155
6.700E-03	\$	ADPLL3-G	\$	17	40	156
5.280E-05	\$	ADV5091C	\$	20	40	157
5.280E-05	\$	ADV5092C	\$	20	40	158
5.280E-05	\$	ADV515-C	\$	20	40	159
5.280E-05	\$	ADV6041C	\$	21	31	160
5.280E-05	\$	ADV6042C	\$	21	31	161
5.280E-05	\$	ADV6043C	\$	21	31	162
3.000E-03	\$	ADV6161A	\$	20	31	163
3.000E-03	\$	ADV6162A	\$	20	31	164
3.000E-03	\$	ADV6163A	\$	20	31	165
2.790E-09	\$	AFV11--C	\$	22	7	166
1.200E-05	\$	AFV5521D	\$	21	31	167
5.280E-05	\$	AFV5525C	\$	20	29	168
5.280E-05	\$	AFV5526C	\$	21	29	169
5.280E-05	\$	AFV5527C	\$	21	29	170
5.280E-05	\$	AGV5013C	\$	22	40	171
5.280E-05	\$	AGV5014C	\$	22	40	172
5.280E-05	\$	AGV5021C	\$	22	43	173
5.280E-05	\$	AGV5022C	\$	22	43	174
5.280E-05	\$	AGV5091C	\$	22	40	175
5.280E-05	\$	AGV5092C	\$	22	40	176
5.280E-05	\$	AGV5162C	\$	22	40	177
5.280E-05	\$	AGV5164C	\$	22	40	178
1.200E-05	\$	AGV8061D	\$	21	1	179

5.280E-05	\$	AGVF506C	\$	22	40	180
5.280E-05	\$	AGVF507C	\$	22	40	181
5.280E-05	\$	AGVF510C	\$	22	40	182
5.280E-05	\$	AGVF511C	\$	22	40	183
5.280E-05	\$	AGVF512C	\$	22	40	184
5.280E-05	\$	AGVF611C	\$	22	29	185
5.280E-05	\$	AGVF612C	\$	22	29	186
5.280E-05	\$	AGVF613C	\$	22	29	187
5.280E-05	\$	AGVR601C	\$	21	29	188
5.280E-05	\$	AGVR602C	\$	21	29	189
5.280E-05	\$	AGVR603C	\$	21	29	190
5.280E-05	\$	AGVR611C	\$	22	29	191
5.280E-05	\$	AGVR612C	\$	22	29	192
5.280E-05	\$	AGVR613C	\$	22	29	193
4.800E-04	\$	AMT8161C	\$	20	1	194
3.400E-03	\$	AMT8171A	\$	20	1	195
4.800E-04	\$	AMT8171C	\$	20	1	196
1.000E+00	\$	AMT8171Q	\$	32	1	197
1.200E-05	\$	AMV181-D	\$	20	5	198
1.200E-05	\$	AMV182-D	\$	20	5	199
1.200E-05	\$	AMV5541D	\$	20	29	200
1.000E-03	\$	AOP2H33V	\$	32	1	201
1.000E-02	\$	AOPHLDPQ	\$	32	1	202
1.600E-04	\$	AOPZZV1V	\$	32	1	203
8.160E-07	\$	APPECCSL	\$	20	1	204
8.160E-07	\$	APPRWS-L	\$	20	1	205
2.640E-05	\$	APV6141C	\$	22	29	206
6.200E-04	\$	ASV5021A	\$	31	40	207
6.200E-04	\$	ASV5022A	\$	31	40	208
6.200E-04	\$	ASV6001A	\$	31	29	209
6.200E-04	\$	ASV6002A	\$	31	29	210
6.200E-04	\$	ASV6003A	\$	31	29	211
0.000E-01	\$	ATKCV31L	\$	24	5	212
0.000E-01	\$	ATKCV32L	\$	24	5	213
0.000E-01	\$	ATKCV33L	\$	24	5	214
0.000E-01	\$	ATKCV34L	\$	24	5	215
0.000E-01	\$	ATKCV35L	\$	24	5	216
0.000E-01	\$	ATKCV36L	\$	24	5	217
0.000E-01	\$	ATKCV37L	\$	24	5	218
0.000E-01	\$	ATKCV38L	\$	24	5	219
0.000E-01	\$	ATKCV41L	\$	24	5	220
0.000E-01	\$	ATKCV42L	\$	24	5	221
0.000E-01	\$	ATKCV43L	\$	24	5	222
0.000E-01	\$	ATKCV44L	\$	24	5	223
0.000E-01	\$	ATKCV45L	\$	24	5	224
0.000E-01	\$	ATKCV46L	\$	24	5	225
0.000E-01	\$	ATKCV47L	\$	24	5	226
0.000E-01	\$	ATKCV48L	\$	24	5	227
2.040E-05	\$	ATKECCSL	\$	38	1	228
2.040E-05	\$	ATKH1EXL	\$	30	29	229
2.040E-05	\$	ATKH2EXL	\$	30	29	230
2.040E-05	\$	ATKH3EXL	\$	30	29	231
2.040E-05	\$	ATKL1EXL	\$	30	40	232
2.040E-05	\$	ATKL2EXL	\$	30	40	233
2.040E-05	\$	ATKL3EXL	\$	30	40	234
2.040E-05	\$	ATKNO1-L	\$	23	1	235
2.040E-05	\$	ATKNO2-L	\$	23	1	236
2.040E-05	\$	ATKNO3-L	\$	23	1	237
3.000E-04	\$	ATT5021A	\$	22	40	238
3.000E-04	\$	ATT5022A	\$	22	40	239

3.000E-04	\$	ATT510-A	\$	22	40	240
3.000E-04	\$	ATT5521A	\$	22	29	241
3.000E-04	\$	ATT5522A	\$	22	29	242
3.000E-04	\$	ATT5523A	\$	22	29	243
5.280E-05	\$	ATV501-C	\$	22	40	244
5.280E-05	\$	ATV502-C	\$	22	40	245
5.280E-05	\$	ATV516-C	\$	22	40	246
5.280E-05	\$	ATV5513C	\$	21	31	247
5.280E-05	\$	ATV5532C	\$	22	31	248
5.280E-05	\$	ATV5533C	\$	22	31	249
5.280E-05	\$	ATV5534C	\$	22	31	250
5.280E-05	\$	ATV5535C	\$	22	31	251
5.280E-05	\$	ATV5583C	\$	22	29	252
5.280E-05	\$	ATV5586C	\$	22	29	253
5.280E-05	\$	ATV5781C	\$	22	29	254
5.280E-05	\$	ATV5782C	\$	22	29	255
5.280E-05	\$	ATV6221C	\$	22	29	256
5.280E-05	\$	ATV8161C	\$	22	1	257
5.280E-05	\$	ATV8162C	\$	22	1	258
5.280E-05	\$	ATV8163C	\$	22	1	259
5.280E-05	\$	ATV8171C	\$	22	1	260
5.280E-05	\$	ATV8181C	\$	22	1	261
5.280E-05	\$	ATV8182C	\$	22	1	262
5.280E-05	\$	AXV1-1-C	\$	21	7	263
5.280E-05	\$	AXV1-2-C	\$	21	7	264
5.280E-05	\$	AXV1-3-C	\$	21	7	265
5.280E-05	\$	AXV1-4-C	\$	21	7	266
5.280E-05	\$	AXV1-5-C	\$	21	7	267
5.280E-05	\$	AXV1-6-C	\$	21	7	268
5.280E-05	\$	AXV1-7-C	\$	21	7	269
5.280E-05	\$	AXV1-8-C	\$	21	7	270
5.280E-05	\$	AXV1-9-C	\$	21	7	271
5.280E-05	\$	AXV110-C	\$	21	7	272
5.280E-05	\$	AXV111-C	\$	21	7	273
5.280E-05	\$	AXV112-C	\$	21	7	274
5.280E-05	\$	AXV113-C	\$	21	7	275
5.280E-05	\$	AXV114-C	\$	21	7	276
5.280E-05	\$	AXV115-C	\$	21	7	277
5.280E-05	\$	AXV116-C	\$	21	7	278
5.280E-05	\$	AXV516-C	\$	22	40	279
5.280E-05	\$	AXV6201C	\$	21	29	280
5.280E-05	\$	AXVBB11C	\$	22	5	281
5.280E-05	\$	AXVBB12C	\$	22	5	282
5.280E-05	\$	AXVBB13C	\$	22	5	283
5.280E-05	\$	AXVBB14C	\$	22	5	284
5.280E-05	\$	AXVBB15C	\$	22	5	285
5.280E-05	\$	AXVBB16C	\$	22	5	286
5.280E-05	\$	AXVBB17C	\$	22	5	287
5.280E-05	\$	AXVBB18C	\$	22	5	288
5.280E-05	\$	AXVBB21C	\$	22	5	289
5.280E-05	\$	AXVBB22C	\$	22	5	290
5.280E-05	\$	AXVBB23C	\$	22	5	291
5.280E-05	\$	AXVBB24C	\$	22	5	292
5.280E-05	\$	AXVBB25C	\$	22	5	293
5.280E-05	\$	AXVBB26C	\$	22	5	294
5.280E-05	\$	AXVBB27C	\$	22	5	295
5.280E-05	\$	AXVBB28C	\$	22	5	296
5.280E-05	\$	AXVBV11C	\$	22	5	297
5.280E-05	\$	AXVBV12C	\$	22	5	298
5.280E-05	\$	AXVBV13C	\$	22	5	299

5.280E-05	\$	AXVBV14C	\$	22	5	300
5.280E-05	\$	AXVBV15C	\$	22	5	301
5.280E-05	\$	AXVBV16C	\$	22	5	302
5.280E-05	\$	AXVBV17C	\$	22	5	303
5.280E-05	\$	AXVBV18C	\$	22	5	304
5.280E-05	\$	AXVBV21C	\$	22	5	305
5.280E-05	\$	AXVBV22C	\$	22	5	306
5.280E-05	\$	AXVBV23C	\$	22	5	307
5.280E-05	\$	AXVBV24C	\$	22	5	308
5.280E-05	\$	AXVBV25C	\$	22	5	309
5.280E-05	\$	AXVBV26C	\$	22	5	310
5.280E-05	\$	AXVBV27C	\$	22	5	311
5.280E-05	\$	AXVBV28C	\$	22	5	312
7.440E-06	\$	CAG3011C	\$	20	18	313
7.440E-06	\$	CAG3081C	\$	20	18	314
7.200E-05	\$	CAN00010	\$	40	8	315
7.200E-05	\$	CAN00020	\$	40	8	316
7.200E-05	\$	CAN00030	\$	40	8	317
7.200E-05	\$	CAN00040	\$	40	8	318
7.200E-05	\$	CAN00050	\$	40	8	319
7.200E-05	\$	CAN00060	\$	40	8	320
7.200E-05	\$	CAN00090	\$	40	8	321
7.200E-05	\$	CAN00100	\$	40	8	322
7.200E-05	\$	CAN00120	\$	40	8	323
7.200E-05	\$	CAN00130	\$	40	8	324
7.200E-05	\$	CAN00140	\$	40	8	325
7.200E-05	\$	CAN00150	\$	40	8	326
7.200E-05	\$	CAN00180	\$	40	8	327
7.200E-05	\$	CAN04220	\$	40	8	328
7.200E-05	\$	CAN04230	\$	40	8	329
7.200E-05	\$	CAN04240	\$	40	8	330
7.200E-05	\$	CAN04250	\$	40	8	331
7.200E-05	\$	CAN04260	\$	40	8	332
7.200E-05	\$	CAN04270	\$	40	8	333
7.200E-05	\$	CAN04280	\$	40	8	334
7.440E-06	\$	CAV2341C	\$	20	18	335
7.440E-06	\$	CAV2342C	\$	20	18	336
1.176E-05	\$	CCV0007C	\$	21	7	337
1.176E-05	\$	CCV2050C	\$	21	14	338
1.176E-05	\$	CCV2060C	\$	22	25	339
2.200E-03	\$	CCV3031B	\$	21	18	340
1.176E-05	\$	CCV3031C	\$	21	18	341
2.200E-03	\$	CCV3032B	\$	21	18	342
1.176E-05	\$	CCV3032C	\$	21	18	343
1.176E-05	\$	CCV3033C	\$	21	18	344
7.440E-06	\$	CDB0001C	\$	20	10	345
3.700E-02	\$	CDB3161A	\$	20	18	346
3.700E-02	\$	CDB3162A	\$	20	14	347
3.700E-02	\$	CDT0011A	\$	20	7	348
1.200E-05	\$	CDT0011D	\$	20	7	349
3.700E-02	\$	CDT0012A	\$	20	9	350
1.200E-05	\$	CDT0012D	\$	20	9	351
3.700E-02	\$	CDV2091A	\$	19	48	352
3.700E-02	\$	CDV2092A	\$	19	48	353
3.700E-02	\$	CDV2101A	\$	19	48	354
3.700E-02	\$	CDV2102A	\$	19	48	355
7.440E-06	\$	CDV3091C	\$	20	18	356
7.440E-06	\$	CDV3092C	\$	20	18	357
1.200E-05	\$	CDV3121D	\$	20	18	358
3.700E-02	\$	CDV3181B	\$	19	18	359

3.700E-02	\$	CDV3182B	\$	19	18	360
4.800E-06	\$	CFC0233N	\$	13	7	361
7.200E-05	\$	CFEBULKO	\$	22	10	362
7.200E-05	\$	CFEBULKO	\$	22	10	363
7.200E-05	\$	CFEVENTO	\$	22	18	364
7.200E-05	\$	CFI0214N	\$	22	7	365
3.000E-04	\$	CFNORIFE	\$	22	7	366
3.000E-08	\$	CFNORIFL	\$	22	7	367
7.200E-05	\$	CFR0242N	\$	13	8	368
7.200E-05	\$	CFR1400M	\$	13	8	369
7.200E-05	\$	CFX0257N	\$	12	18	370
7.200E-05	\$	CFX0FT7M	\$	12	18	371
7.200E-05	\$	CFX0T66M	\$	13	7	372
7.200E-05	\$	CFX1109M	\$	13	7	373
7.200E-05	\$	CFX1110M	\$	13	7	374
1.248E-06	\$	CGV0075C	\$	22	7	375
1.248E-06	\$	CGV0077C	\$	22	7	376
1.248E-06	\$	CGV2031C	\$	20	14	377
1.248E-06	\$	CGV2032C	\$	20	14	378
1.248E-06	\$	CGV2033C	\$	20	14	379
1.248E-06	\$	CGV3061C	\$	19	18	380
1.248E-06	\$	CGV3081C	\$	19	48	381
1.248E-06	\$	CGV3082C	\$	19	48	382
1.248E-06	\$	CGV3121C	\$	19	48	383
1.248E-06	\$	CGV3122C	\$	19	48	384
4.800E-04	\$	CHX0001L	\$	24	14	385
4.800E-04	\$	CHX0002L	\$	24	14	386
4.800E-04	\$	CHX0003L	\$	24	14	387
4.800E-04	\$	CHX0004L	\$	24	14	388
4.800E-06	\$	CLC0233M	\$	13	8	389
7.200E-05	\$	CLR0205M	\$	13	8	390
7.200E-05	\$	CLX0201M	\$	13	7	391
7.200E-05	\$	CLX0257M	\$	13	7	392
6.960E-06	\$	CMB20R1C	\$	20	18	393
6.960E-06	\$	CMB20R2C	\$	20	18	394
6.960E-06	\$	CMB20R3C	\$	20	18	395
6.960E-06	\$	CMB20R4C	\$	20	18	396
6.960E-06	\$	CMB20S1C	\$	20	18	397
6.960E-06	\$	CMB20S2C	\$	20	18	398
6.960E-06	\$	CMB20S3C	\$	20	18	399
6.960E-06	\$	CMB20S4C	\$	20	18	400
2.400E-02	\$	CMG3091A	\$	19	49	401
2.400E-02	\$	CMG3099A	\$	19	48	402
4.800E-05	\$	CMP0001G	\$	16	17	403
4.800E-05	\$	CMP0002G	\$	32	17	404
9.100E-02	\$	CMP0003F	\$	16	17	405
9.800E-04	\$	CMP0003G	\$	32	17	406
1.920E-06	\$	CMP0003W	\$	32	17	407
1.920E-06	\$	CMP0003X	\$	32	17	408
6.960E-06	\$	CMT2041C	\$	20	18	409
6.960E-06	\$	CMT3011C	\$	20	14	410
6.960E-06	\$	CMT3012C	\$	20	14	411
6.960E-06	\$	CMT3013C	\$	20	14	412
6.960E-06	\$	CMT3021C	\$	20	18	413
6.960E-06	\$	CMT3022C	\$	20	18	414
6.960E-06	\$	CMT3023C	\$	20	18	415
6.960E-06	\$	CMT3024C	\$	20	18	416
6.960E-06	\$	CMT302RC	\$	20	18	417
6.960E-06	\$	CMT3031C	\$	20	14	418
6.960E-06	\$	CMT3032C	\$	20	14	419

6.960E-06	\$	CMT3033C	\$	20	14	420
6.960E-06	\$	CMT3034C	\$	20	14	421
1.200E-05	\$	CMT3035D	\$	20	18	422
6.960E-06	\$	CMT3041C	\$	20	18	423
6.960E-06	\$	CMT3042C	\$	20	18	424
2.400E-02	\$	CMT3043A	\$	20	18	425
6.960E-06	\$	CMT3043C	\$	20	18	426
1.200E-05	\$	CMT3071D	\$	20	18	427
2.400E-02	\$	CMT3151B	\$	19	18	428
2.400E-02	\$	CMT3152B	\$	19	18	429
6.960E-06	\$	CMT3171C	\$	19	18	430
6.960E-06	\$	CMT3172C	\$	19	18	431
6.960E-06	\$	CMV2071C	\$	19	49	432
1.000E+00	\$	COPBYPAR	\$	32	1	433
0.000E-01	\$	COPBYPSQ	\$	32	7	434
0.000E-01	\$	COPFLNGV	\$	32	7	435
1.000E-04	\$	COPFLTRU	\$	32	7	436
1.000E+00	\$	COPMP3Q	\$	32	7	437
1.000E+00	\$	COPSBUSQ	\$	32	7	438
1.000E-02	\$	COPSWITQ	\$	32	7	439
1.000E-04	\$	COPTETRU	\$	32	7	440
1.000E-02	\$	COPTRIPO	\$	32	7	441
4.800E-06	\$	CPC02450	\$	13	8	442
4.800E-06	\$	CPC02490	\$	13	8	443
7.200E-05	\$	CPG11080	\$	13	7	444
8.160E-07	\$	CPP105NL	\$	32	7	445
8.160E-07	\$	CPP109NL	\$	32	18	446
8.160E-07	\$	CPPBILCL	\$	32	7	447
8.160E-07	\$	CPPDUMPL	\$	32	9	448
7.200E-05	\$	CPR02100	\$	12	8	449
1.000E-04	\$	CPRESS	\$	32	10	450
1.000E-04	\$	CPS02010	\$	13	14	451
1.000E-04	\$	CPS02020	\$	13	14	452
1.000E-04	\$	CPS02300	\$	13	7	453
1.000E-04	\$	CPS02310	\$	13	7	454
1.000E-04	\$	CPS03250	\$	13	7	455
1.000E-04	\$	CPS04000	\$	13	7	456
8.160E-05	\$	CPV0076C	\$	19	7	457
1.600E-02	\$	CPVGN27A	\$	19	7	458
7.200E-05	\$	CPX0201N	\$	22	14	459
7.200E-05	\$	CPX0202N	\$	22	14	460
7.200E-05	\$	CPX02040	\$	22	18	461
7.200E-05	\$	CPX0205N	\$	13	18	462
7.200E-05	\$	CPX02250	\$	22	18	463
2.300E-02	\$	CRV0002A	\$	21	7	464
4.800E-05	\$	CRV0002D	\$	21	7	465
2.300E-02	\$	CRV3051A	\$	22	49	466
2.300E-02	\$	CRV3071A	\$	22	49	467
2.300E-02	\$	CRV3111A	\$	22	18	468
1.100E-03	\$	CSV0246A	\$	19	18	469
1.100E-03	\$	CSV0247A	\$	19	18	470
1.000E-04	\$	CSW00410	\$	13	7	471
4.800E-06	\$	CTC0261N	\$	13	8	472
7.200E-05	\$	CTE11120	\$	13	8	473
7.200E-05	\$	CTI0399N	\$	13	7	474
7.200E-05	\$	CTI0T660	\$	13	7	475
2.040E-05	\$	CTK0000L	\$	24	49	476
2.040E-05	\$	CTKLPN2L	\$	23	7	477
2.040E-05	\$	CTKLQN2L	\$	23	7	478
7.200E-05	\$	CTM11120	\$	13	8	479

7.200E-05	\$	CTMFLOWO	\$	13	8	480
7.200E-05	\$	CTR0256N	\$	13	8	481
1.000E-04	\$	CTS02800	\$	13	7	482
1.248E-06	\$	CTV2200C	\$	19	48	483
1.248E-06	\$	CTV2300C	\$	20	18	484
1.200E-05	\$	CTV3190D	\$	20	22	485
6.200E-03	\$	CTV3191B	\$	19	18	486
6.200E-03	\$	CTV3192B	\$	19	18	487
1.248E-06	\$	CTV3201C	\$	19	18	488
1.248E-06	\$	CTV3202C	\$	19	18	489
1.248E-06	\$	CTVGH51C	\$	22	7	490
1.248E-06	\$	CTVGN26C	\$	22	7	491
7.200E-05	\$	CTX0201N	\$	22	18	492
7.200E-05	\$	CTX0249N	\$	13	7	493
7.200E-05	\$	CTX0250N	\$	13	7	494
7.200E-05	\$	CTX0251N	\$	13	7	495
7.200E-05	\$	CTX0252N	\$	13	7	496
7.200E-05	\$	CTX0269N	\$	13	7	497
7.200E-05	\$	CTX0270N	\$	13	7	498
7.200E-05	\$	CTX0271N	\$	13	7	499
7.200E-05	\$	CTX0272N	\$	13	7	500
7.200E-05	\$	CTX0284N	\$	13	7	501
7.200E-05	\$	CTX0285N	\$	13	7	502
7.200E-05	\$	CTX0286N	\$	13	7	503
1.248E-06	\$	CXV3126C	\$	22	14	504
1.248E-06	\$	CXV3128C	\$	22	14	505
9.899E-01	\$	CZZGCSDP	\$	32	7	506
4.500E-05	\$	DDA15	\$	12	5	507
1.100E-08	\$	DDA25	\$	12	5	508
1.047E-03	\$	DDA43	\$	12	5	509
1.205E-04	\$	EAB2011C	\$	20	28	510
1.205E-04	\$	EAB2013C	\$	20	28	511
1.205E-04	\$	EAB2015C	\$	20	28	512
1.205E-04	\$	EAB2017C	\$	20	28	513
1.205E-04	\$	EAB2019C	\$	20	28	514
1.205E-04	\$	EAB2021C	\$	20	28	515
1.205E-04	\$	EAB2023C	\$	20	28	516
1.205E-04	\$	EAB2025C	\$	20	28	517
1.205E-04	\$	EAB2027C	\$	20	28	518
1.205E-04	\$	EAB2029C	\$	20	28	519
1.205E-04	\$	EAB2031C	\$	20	28	520
1.205E-04	\$	EAB2033C	\$	20	28	521
1.205E-04	\$	EAB2035C	\$	20	28	522
1.205E-04	\$	EAB2037C	\$	20	28	523
1.205E-04	\$	EAB2039C	\$	20	28	524
1.205E-04	\$	EAB2221C	\$	20	28	525
1.205E-04	\$	EAB2223C	\$	20	28	526
1.205E-04	\$	EAB2225C	\$	20	28	527
1.205E-04	\$	EAB2227C	\$	20	28	528
1.205E-04	\$	EAB2229C	\$	20	28	529
1.480E-03	\$	EAV1101B	\$	22	28	530
1.480E-03	\$	EAV1102B	\$	22	28	531
1.480E-03	\$	EAV1103B	\$	22	28	532
1.205E-04	\$	EAV2060C	\$	20	23	533
1.205E-04	\$	EAV2061C	\$	20	23	534
1.205E-04	\$	EAV2062C	\$	20	23	535
1.205E-04	\$	EAV2063C	\$	20	23	536
1.205E-04	\$	EAV2064C	\$	20	23	537
1.205E-04	\$	EAV2065C	\$	20	23	538
1.205E-04	\$	EAV2066C	\$	20	23	539

1.205E-04	\$	EAV2067C	\$	20	23	540
1.205E-04	\$	EAV2068C	\$	20	23	541
1.205E-04	\$	EAV2069C	\$	20	23	542
1.150E-03	\$	ECCDAASP	\$	12	5	543
7.200E-05	\$	ECM2351P	\$	13	25	544
7.200E-05	\$	ECM2352P	\$	13	25	545
7.200E-05	\$	ECM2353P	\$	13	25	546
7.200E-05	\$	ECM2354P	\$	13	25	547
7.200E-05	\$	ECM2355P	\$	13	25	548
7.200E-05	\$	ECMDT1GP	\$	12	25	549
7.200E-05	\$	ECMDT2GP	\$	12	25	550
7.200E-05	\$	ECMDT3GP	\$	12	25	551
7.200E-05	\$	ECMDT4GP	\$	12	25	552
7.200E-05	\$	ECMDT5GP	\$	12	25	553
7.200E-05	\$	ECMDTHPP	\$	13	28	554
7.200E-05	\$	ECMDTLCP	\$	13	28	555
7.200E-05	\$	ECMSGL0P	\$	13	14	556
7.200E-05	\$	ECMSGL1P	\$	13	14	557
7.200E-05	\$	ECMSGL2P	\$	13	14	558
7.200E-05	\$	ECMSGL3P	\$	13	14	559
7.200E-05	\$	ECMSGL4P	\$	13	14	560
7.200E-05	\$	ECMSGL5P	\$	13	14	561
7.200E-05	\$	ECMSGL6P	\$	13	14	562
7.200E-05	\$	ECMSGL7P	\$	13	14	563
7.200E-05	\$	ECMSGL8P	\$	13	14	564
7.200E-05	\$	ECMSGL9P	\$	13	14	565
7.200E-05	\$	ECMSTLCP	\$	13	27	566
3.300E-05	\$	ECV1041A	\$	21	46	567
3.300E-05	\$	ECV1042A	\$	21	46	568
3.300E-05	\$	ECV1043A	\$	21	46	569
3.300E-05	\$	ECV1121A	\$	21	51	570
3.300E-05	\$	ECV2420A	\$	21	23	571
3.300E-05	\$	ECV2422A	\$	21	23	572
3.300E-05	\$	ECV2424A	\$	21	23	573
3.300E-05	\$	ECV2426A	\$	21	23	574
3.300E-05	\$	ECV2428A	\$	21	23	575
1.480E-03	\$	EDT2431A	\$	20	27	576
1.205E-04	\$	EDT2431C	\$	20	27	577
1.480E-03	\$	EDV1061A	\$	20	51	578
1.200E-05	\$	EDV1061D	\$	20	51	579
1.480E-03	\$	EDV1141A	\$	20	51	580
1.205E-04	\$	EDV1141C	\$	20	51	581
1.480E-03	\$	EDV2031A	\$	20	23	582
1.205E-04	\$	EDV2031C	\$	20	23	583
1.480E-03	\$	EDV2032A	\$	20	27	584
1.205E-04	\$	EDV2032C	\$	20	27	585
1.200E-05	\$	EDV2151D	\$	21	14	586
1.200E-05	\$	EDV2152D	\$	21	14	587
1.205E-04	\$	EDV2160C	\$	20	23	588
1.205E-04	\$	EDV2161C	\$	20	23	589
1.205E-04	\$	EDV2162C	\$	20	23	590
1.205E-04	\$	EDV2163C	\$	20	23	591
1.205E-04	\$	EDV2164C	\$	20	23	592
1.205E-04	\$	EDV2165C	\$	20	23	593
1.205E-04	\$	EDV2166C	\$	20	23	594
1.205E-04	\$	EDV2167C	\$	20	23	595
1.205E-04	\$	EDV2168C	\$	20	23	596
1.205E-04	\$	EDV2169C	\$	20	23	597
1.200E-05	\$	EDV2351D	\$	20	23	598
1.200E-05	\$	EDV2352D	\$	20	23	599

1.200E-05	\$	EDV2353D	\$	20	23	600
1.200E-05	\$	EDV2354D	\$	20	23	601
1.200E-05	\$	EDV2355D	\$	20	23	602
1.205E-04	\$	EDV6211C	\$	20	28	603
1.205E-04	\$	EDV6212C	\$	20	28	604
1.205E-04	\$	EDV6213C	\$	20	28	605
1.205E-04	\$	EDV6214C	\$	20	28	606
2.472E-05	\$	EFV1213C	\$	21	23	607
2.472E-05	\$	EFV2131C	\$	21	23	608
2.472E-05	\$	EFV2132C	\$	21	23	609
2.472E-05	\$	EFV2133C	\$	21	23	610
2.472E-05	\$	EFV2134C	\$	21	23	611
2.472E-05	\$	EFV2135C	\$	21	23	612
2.472E-05	\$	EFV2213C	\$	21	23	613
2.472E-05	\$	EFV3213C	\$	21	23	614
2.472E-05	\$	EFV4213C	\$	21	23	615
2.472E-05	\$	EFV5213C	\$	21	23	616
1.286E-05	\$	EGV2071C	\$	21	28	617
1.286E-05	\$	EGV2072C	\$	21	28	618
1.286E-05	\$	EGV2073C	\$	21	28	619
1.286E-05	\$	EGV2074C	\$	21	28	620
1.286E-05	\$	EGV2411C	\$	19	23	621
1.286E-05	\$	EGV2412C	\$	19	23	622
1.286E-05	\$	EGV2413C	\$	19	23	623
1.286E-05	\$	EGV2414C	\$	19	23	624
1.286E-05	\$	EGV2416C	\$	19	23	625
1.286E-05	\$	EGV2417C	\$	19	23	626
1.286E-05	\$	EGV2418C	\$	19	23	627
1.286E-05	\$	EGV2419C	\$	19	23	628
1.286E-05	\$	EGV4111C	\$	19	23	629
1.286E-05	\$	EGV4112C	\$	19	23	630
1.286E-05	\$	EGV4113C	\$	19	23	631
1.286E-05	\$	EGV4114C	\$	19	23	632
1.286E-05	\$	EGV4116C	\$	19	23	633
1.286E-05	\$	EGV4117C	\$	19	23	634
1.286E-05	\$	EGV4118C	\$	19	23	635
1.286E-05	\$	EGV4119C	\$	19	23	636
1.286E-05	\$	EGV4121C	\$	19	23	637
1.286E-05	\$	EGV4122C	\$	19	23	638
1.286E-05	\$	EGV4123C	\$	19	23	639
1.286E-05	\$	EGV4124C	\$	19	23	640
2.010E-04	\$	EMPDT1AG	\$	32	24	641
2.010E-04	\$	EMPDT1BG	\$	32	24	642
2.010E-04	\$	EMPDT2AG	\$	32	24	643
2.010E-04	\$	EMPDT2BG	\$	32	24	644
2.010E-04	\$	EMPDT3AG	\$	32	24	645
2.010E-04	\$	EMPDT3BG	\$	32	24	646
2.010E-04	\$	EMPDT4AG	\$	32	24	647
2.010E-04	\$	EMPDT4BG	\$	32	24	648
2.010E-04	\$	EMPDT5AG	\$	32	24	649
2.010E-04	\$	EMPDT5BG	\$	32	24	650
5.540E-05	\$	EMPFWP1G	\$	32	47	651
5.540E-05	\$	EMPFWP2G	\$	32	47	652
5.540E-05	\$	EMPFWP3G	\$	32	47	653
2.496E-04	\$	EMT1031C	\$	20	46	654
2.496E-04	\$	EMT1032C	\$	20	46	655
2.496E-04	\$	EMT1033C	\$	20	46	656
3.000E-03	\$	EMV2041A	\$	20	14	657
3.000E-03	\$	EMV2042A	\$	20	14	658
3.000E-03	\$	EMV2043A	\$	20	14	659

3.000E-03	\$	EMV2044A	\$	20	14	660
3.000E-03	\$	EMV2045A	\$	20	14	661
1.000E-02	\$	EOPTGOVU	\$	32	1	662
8.160E-07	\$	EPP1M-CL	\$	32	1	663
8.160E-07	\$	EPP28HPL	\$	32	1	664
8.160E-07	\$	EPP2C1AL	\$	32	1	665
8.160E-07	\$	EPP2C1BL	\$	32	1	666
8.160E-07	\$	EPP2C2AL	\$	32	1	667
8.160E-07	\$	EPP2C2BL	\$	32	1	668
8.160E-07	\$	EPP2C3AL	\$	32	1	669
8.160E-07	\$	EPP2C3BL	\$	32	1	670
8.160E-07	\$	EPP2C4AL	\$	32	1	671
8.160E-07	\$	EPP2C4BL	\$	32	1	672
8.160E-07	\$	EPP2C5AL	\$	32	1	673
8.160E-07	\$	EPP2C5BL	\$	32	1	674
8.160E-07	\$	EPP2M-CL	\$	32	1	675
8.160E-07	\$	EPPAHRPL	\$	32	1	676
8.160E-07	\$	EPPDEC2L	\$	32	1	677
8.160E-07	\$	EPPFWPHL	\$	32	1	678
8.160E-07	\$	EPPIBH-L	\$	32	1	679
8.160E-07	\$	EPPMSDTL	\$	32	1	680
8.160E-07	\$	EPPMSH-L	\$	32	1	681
8.160E-07	\$	EPPPSDEL	\$	32	1	682
8.160E-07	\$	EPPSLMUL	\$	32	1	683
1.000E+00	\$	ESCTURBP	\$	32	1	684
1.000E-04	\$	ESTDT-1G	\$	28	25	685
1.000E-04	\$	ESTDT-1G	\$	28	25	686
1.000E-04	\$	ESTDT-2G	\$	28	25	687
1.000E-04	\$	ESTDT-2G	\$	28	25	688
1.000E-04	\$	ESTDT-3G	\$	28	25	689
1.000E-04	\$	ESTDT-3G	\$	28	25	690
1.000E-04	\$	ESTDT-4G	\$	28	25	691
1.000E-04	\$	ESTDT-4G	\$	28	25	692
1.000E-04	\$	ESTDT-5G	\$	28	25	693
1.000E-04	\$	ESTDT-5G	\$	28	25	694
8.160E-07	\$	ETKDESTL	\$	23	51	695
1.286E-05	\$	ETV1011C	\$	21	51	696
1.286E-05	\$	ETV1012C	\$	21	51	697
1.286E-05	\$	ETV1013C	\$	21	51	698
1.286E-05	\$	ETV1061C	\$	21	51	699
1.286E-05	\$	ETV1104C	\$	21	51	700
1.286E-05	\$	ETV1106C	\$	22	46	701
1.286E-05	\$	ETV1111C	\$	22	46	702
1.286E-05	\$	ETV1112C	\$	22	46	703
1.286E-05	\$	ETV1113C	\$	22	46	704
1.286E-05	\$	ETV1171D	\$	22	51	705
1.286E-05	\$	ETV2010C	\$	20	28	706
1.286E-05	\$	ETV2012C	\$	20	28	707
1.286E-05	\$	ETV2014C	\$	20	28	708
1.286E-05	\$	ETV2016C	\$	20	28	709
1.286E-05	\$	ETV2018C	\$	20	28	710
1.286E-05	\$	ETV2020C	\$	20	28	711
1.286E-05	\$	ETV2022C	\$	20	28	712
1.286E-05	\$	ETV2024C	\$	20	28	713
1.286E-05	\$	ETV2026C	\$	20	28	714
1.286E-05	\$	ETV2028C	\$	20	28	715
1.286E-05	\$	ETV2030C	\$	20	28	716
1.286E-05	\$	ETV2032C	\$	20	28	717
1.286E-05	\$	ETV2034C	\$	20	28	718
1.286E-05	\$	ETV2036C	\$	20	28	719

1.286E-05	\$	ETV2038C	\$	20	28	720
1.286E-05	\$	ETV2040C	\$	20	28	721
1.286E-05	\$	ETV2052C	\$	20	25	722
1.286E-05	\$	ETV2104C	\$	21	51	723
1.286E-05	\$	ETV2106C	\$	22	46	724
1.286E-05	\$	ETV2222C	\$	20	28	725
1.286E-05	\$	ETV2224C	\$	20	28	726
1.286E-05	\$	ETV2226C	\$	20	28	727
1.286E-05	\$	ETV2228C	\$	20	28	728
1.286E-05	\$	ETV2340C	\$	20	23	729
1.286E-05	\$	ETV2341C	\$	20	23	730
1.286E-05	\$	ETV2342C	\$	20	23	731
1.286E-05	\$	ETV2343C	\$	20	23	732
1.286E-05	\$	ETV2344C	\$	20	23	733
1.286E-05	\$	ETV2345C	\$	20	23	734
1.286E-05	\$	ETV2346C	\$	20	23	735
1.286E-05	\$	ETV2347C	\$	20	23	736
1.286E-05	\$	ETV2348C	\$	20	23	737
1.286E-05	\$	ETV2349C	\$	20	23	738
1.286E-05	\$	ETV2351C	\$	20	25	739
1.286E-05	\$	ETV2352C	\$	20	25	740
1.286E-05	\$	ETV2353C	\$	20	25	741
1.286E-05	\$	ETV2354C	\$	20	25	742
1.286E-05	\$	ETV2355C	\$	20	25	743
1.286E-05	\$	ETV2356C	\$	20	27	744
1.286E-05	\$	ETV2380C	\$	20	23	745
1.286E-05	\$	ETV2381C	\$	20	23	746
1.286E-05	\$	ETV2382C	\$	20	23	747
1.286E-05	\$	ETV2383C	\$	20	23	748
1.286E-05	\$	ETV2384C	\$	20	23	749
1.286E-05	\$	ETV2385C	\$	20	23	750
1.286E-05	\$	ETV2386C	\$	20	23	751
1.286E-05	\$	ETV2387C	\$	20	23	752
1.286E-05	\$	ETV2388C	\$	20	23	753
1.286E-05	\$	ETV2389C	\$	20	23	754
1.286E-05	\$	ETV2441C	\$	20	27	755
1.286E-05	\$	ETV2442C	\$	20	27	756
1.286E-05	\$	ETV307-C	\$	20	27	757
1.286E-05	\$	ETV308-C	\$	20	27	758
1.286E-05	\$	ETV3106C	\$	22	46	759
1.286E-05	\$	ETV3411C	\$	20	23	760
1.286E-05	\$	ETV3412C	\$	20	23	761
1.286E-05	\$	ETV3413C	\$	20	23	762
1.286E-05	\$	ETV3414C	\$	20	23	763
1.286E-05	\$	ETV3415C	\$	20	23	764
1.286E-05	\$	ETV3416C	\$	20	23	765
1.286E-05	\$	ETV3417C	\$	20	23	766
1.286E-05	\$	ETV3418C	\$	20	23	767
1.286E-05	\$	ETV3419C	\$	20	23	768
1.286E-05	\$	ETV3420C	\$	20	23	769
1.286E-05	\$	ETV3422C	\$	20	23	770
1.286E-05	\$	ETV3424C	\$	20	23	771
1.286E-05	\$	ETV3426C	\$	20	23	772
1.286E-05	\$	ETV3428C	\$	20	23	773
1.286E-05	\$	ETV3430C	\$	20	23	774
1.286E-05	\$	ETV4104C	\$	21	51	775
3.000E-03	\$	FCBHPASK	\$	3	31	776
3.000E-03	\$	FCBHPBSK	\$	3	31	777
4.800E-06	\$	FCMSC2UP	\$	12	7	778
4.800E-06	\$	FCMSC2VP	\$	12	7	779

4.800E-06	\$	FCMSC2WP	\$	12	7	780
4.800E-06	\$	FCMSC2XP	\$	12	7	781
3.840E-05	\$	FCV551UC	\$	21	29	782
3.840E-05	\$	FCV551VC	\$	21	29	783
4.500E-04	\$	FCV551WA	\$	21	29	784
4.500E-04	\$	FCV551XA	\$	21	29	785
3.840E-05	\$	FCV579UC	\$	22	29	786
3.840E-05	\$	FCV579WC	\$	22	29	787
4.500E-04	\$	FCVB79UA	\$	22	29	788
4.500E-04	\$	FCVB79WA	\$	22	29	789
7.200E-05	\$	FFDHPAOP	\$	15	29	790
7.200E-05	\$	FFDHPASP	\$	15	29	791
7.200E-05	\$	FFDHPBOP	\$	15	29	792
7.200E-05	\$	FFDHPBSP	\$	15	29	793
7.200E-05	\$	FFE290BO	\$	10	31	794
2.040E-05	\$	FFV5511C	\$	13	29	795
2.040E-05	\$	FFV5512C	\$	21	29	796
2.040E-05	\$	FFV552UC	\$	21	29	797
2.040E-05	\$	FFV552VC	\$	21	29	798
2.040E-05	\$	FFV552WC	\$	21	29	799
2.040E-05	\$	FFV552XC	\$	21	29	800
2.040E-05	\$	FFV553UC	\$	21	29	801
2.040E-05	\$	FFV553VC	\$	21	29	802
2.040E-05	\$	FFV553WC	\$	21	29	803
2.040E-05	\$	FFV553XC	\$	21	29	804
2.040E-05	\$	FFV8031C	\$	21	1	805
7.200E-05	\$	FFX5511N	\$	13	31	806
1.896E-05	\$	FGV5583C	\$	22	31	807
1.896E-05	\$	FGV5593C	\$	22	31	808
4.800E-04	\$	FHXFLCUL	\$	24	29	809
4.800E-04	\$	FHXFLCVL	\$	24	29	810
4.800E-04	\$	FHXFLCWL	\$	24	29	811
4.800E-04	\$	FHXFLCXL	\$	24	29	812
4.800E-04	\$	FHXLBOUL	\$	24	29	813
4.800E-04	\$	FHXLBOVL	\$	24	29	814
4.800E-04	\$	FHXLBOWL	\$	24	29	815
4.800E-04	\$	FHXLBOXL	\$	24	29	816
7.900E-04	\$	FMPAOPUF	\$	16	29	817
5.900E-03	\$	FMPAOPUG	\$	32	29	818
7.900E-04	\$	FMPAOPVF	\$	16	29	819
5.900E-03	\$	FMPAOPVG	\$	32	29	820
7.900E-04	\$	FMPAOPWF	\$	16	29	821
5.900E-03	\$	FMPAOPWG	\$	32	29	822
7.900E-04	\$	FMPAOPXF	\$	16	29	823
5.900E-03	\$	FMPAOPXG	\$	32	29	824
3.840E-05	\$	FMPHPAOG	\$	32	29	825
9.100E-05	\$	FMPHPASF	\$	16	29	826
3.840E-05	\$	FMPHPASC	\$	32	29	827
3.840E-05	\$	FMPHPBOG	\$	32	29	828
9.100E-05	\$	FMPHPBSF	\$	16	29	829
3.840E-05	\$	FMPHPBSG	\$	32	29	830
3.840E-05	\$	FMPOLPUG	\$	32	29	831
3.840E-05	\$	FMPOLPVG	\$	32	29	832
3.840E-05	\$	FMPOLPWG	\$	32	29	833
3.840E-05	\$	FMPOLPXG	\$	32	29	834
1.000E-04	\$	FOP5511U	\$	32	1	835
1.000E-02	\$	FOPHPDAV	\$	32	1	836
1.000E-04	\$	FOPHPIPU	\$	32	1	837
1.000E+00	\$	FOPHPsBR	\$	32	1	838
1.000E-03	\$	FOPLSCUV	\$	32	1	839

1.000E-03	\$	FOPPLSCVV	\$	32	1	840
1.000E-03	\$	FOPPLSCWV	\$	32	1	841
1.000E-03	\$	FOPPLSCXV	\$	32	1	842
1.000E-03	\$	FOPRSCUV	\$	32	1	843
1.000E-03	\$	FOPRSCVV	\$	32	1	844
1.000E-03	\$	FOPRSCWV	\$	32	1	845
1.000E-03	\$	FOPRSCXV	\$	32	1	846
8.160E-07	\$	FPPHPIPL	\$	32	1	847
7.500E-04	\$	FPV0610A	\$	22	31	848
7.500E-04	\$	FPV5621A	\$	22	31	849
7.200E-05	\$	FPX0260N	\$	13	14	850
7.200E-05	\$	FPX0261N	\$	13	14	851
7.200E-05	\$	FPXHPIPN	\$	13	29	852
1.896E-05	\$	FSV561WC	\$	31	29	853
1.896E-05	\$	FSV561XC	\$	31	29	854
1.000E-04	\$	FSWHPIPO	\$	10	31	855
2.040E-05	\$	FTKOLRUL	\$	24	29	856
2.040E-05	\$	FTKOLRVL	\$	24	29	857
2.040E-05	\$	FTKOLRWL	\$	24	29	858
2.040E-05	\$	FTKOLRXL	\$	24	29	859
7.200E-05	\$	FTMFLCWP	\$	9	31	860
7.200E-05	\$	FTMFLCXP	\$	9	31	861
8.160E-05	\$	FTTFLCUC	\$	22	29	862
8.160E-05	\$	FTTFLCVC	\$	22	29	863
8.160E-05	\$	FTTFLCWC	\$	22	29	864
8.160E-05	\$	FTTFLCXC	\$	22	29	865
8.160E-05	\$	FTTLBOUC	\$	22	29	866
8.160E-05	\$	FTTLBOVC	\$	22	29	867
8.160E-05	\$	FTTLBOWC	\$	22	29	868
8.160E-05	\$	FTTLBOXC	\$	22	29	869
1.896E-05	\$	FTV0611C	\$	22	31	870
1.896E-05	\$	FTV0683C	\$	22	31	871
1.896E-05	\$	FTV0685C	\$	22	31	872
1.896E-05	\$	FTV5513C	\$	22	31	873
1.896E-05	\$	FTV577UC	\$	22	31	874
1.896E-05	\$	FTV577WC	\$	22	31	875
1.896E-05	\$	FTV599UC	\$	22	31	876
1.896E-05	\$	FTV600WC	\$	22	31	877
1.896E-05	\$	FTV602UC	\$	22	29	878
1.896E-05	\$	FTV602VC	\$	22	29	879
1.896E-05	\$	FTV602WC	\$	22	29	880
1.896E-05	\$	FTV602XC	\$	22	29	881
1.896E-05	\$	FTV603UC	\$	22	29	882
1.896E-05	\$	FTV603VC	\$	22	29	883
1.896E-05	\$	FTV603WC	\$	22	29	884
1.896E-05	\$	FTV603XC	\$	22	29	885
1.896E-05	\$	FTVC77UC	\$	22	29	886
1.896E-05	\$	FTVC77WC	\$	22	29	887
1.200E-05	\$	GAG2191D	\$	20	14	888
1.200E-05	\$	GAG2192D	\$	20	14	889
1.480E-03	\$	GAG2531A	\$	20	27	890
1.480E-03	\$	GAG2532A	\$	20	27	891
1.000E+00	\$	GBTBAILK	\$	4	5	892
2.400E-07	\$	GBUACCRI	\$	12	6	893
2.400E-07	\$	GBUACEAI	\$	12	6	894
2.400E-07	\$	GBUIP-1I	\$	12	6	895
2.400E-07	\$	GBUIP-2I	\$	12	6	896
2.400E-07	\$	GBUMAXAI	\$	12	6	897
2.400E-07	\$	GBUMAXBI	\$	12	6	898
7.200E-05	\$	GCB-165H	\$	3	5	899

7.200E-05	\$	GCB3-65H	\$	3	6	900
7.200E-05	\$	GCBBR-CH	\$	4	5	901
7.200E-05	\$	GCBD410H	\$	3	30	902
7.200E-05	\$	GCBD432H	\$	3	30	903
7.200E-05	\$	GCBF1X7H	\$	3	6	904
7.200E-05	\$	GCBF1X8H	\$	3	6	905
7.200E-05	\$	GCBIP-1H	\$	3	5	906
7.200E-05	\$	GCBIP-2H	\$	3	5	907
3.000E-03	\$	GCD1521H	\$	3	31	908
3.000E-03	\$	GCD1522H	\$	3	31	909
3.000E-03	\$	GCD1523H	\$	3	31	910
3.000E-03	\$	GCD1524H	\$	3	31	911
3.000E-03	\$	GCD1525H	\$	3	31	912
7.200E-05	\$	GCMPPC-P	\$	13	14	913
1.000E-04	\$	GCN1521B	\$	3	31	914
1.000E-04	\$	GCN1522B	\$	3	31	915
1.000E-04	\$	GCN1523B	\$	3	31	916
1.000E-04	\$	GCN1524B	\$	3	31	917
1.000E-04	\$	GCN1525B	\$	3	31	918
1.000E-04	\$	GCN2741B	\$	3	31	919
1.000E-04	\$	GCN2742B	\$	3	31	920
1.000E-04	\$	GCN2743B	\$	3	31	921
1.000E-04	\$	GCN2744B	\$	3	31	922
1.000E-04	\$	GCN2745B	\$	3	31	923
1.000E-04	\$	GCNTGN1B	\$	10	26	924
1.000E-04	\$	GCNTGN2B	\$	10	26	925
1.000E-04	\$	GCNTGN3B	\$	10	26	926
1.000E-04	\$	GCNTGN4B	\$	10	26	927
1.000E-04	\$	GCNTGN5B	\$	10	26	928
1.000E-04	\$	GCNTT11B	\$	10	26	929
1.000E-04	\$	GCNTT12B	\$	10	26	930
1.000E-04	\$	GCNTT13B	\$	10	26	931
1.000E-04	\$	GCNTT14B	\$	10	26	932
1.000E-04	\$	GCNTT15B	\$	10	26	933
7.200E-04	\$	GCP1LB1G	\$	32	21	934
7.200E-04	\$	GCP1LB2G	\$	32	21	935
7.200E-04	\$	GCP2LB1G	\$	32	21	936
7.200E-04	\$	GCP2LB2G	\$	32	21	937
7.200E-04	\$	GCP3LB1G	\$	32	21	938
7.200E-04	\$	GCP3LB2G	\$	32	21	939
7.200E-04	\$	GCP4LB1G	\$	32	21	940
7.200E-04	\$	GCP4LB2G	\$	32	21	941
7.200E-04	\$	GCP5LB1G	\$	32	21	942
7.200E-04	\$	GCP5LB2G	\$	32	21	943
1.200E-05	\$	GDV2031D	\$	19	49	944
1.200E-05	\$	GDV2161D	\$	19	49	945
1.200E-05	\$	GDV2171D	\$	19	49	946
7.200E-05	\$	GEP202-M	\$	13	49	947
7.200E-05	\$	GEP2021M	\$	13	49	948
7.200E-05	\$	GEP203-M	\$	13	49	949
7.200E-05	\$	GEP2031M	\$	13	49	950
7.200E-05	\$	GEP244-M	\$	13	49	951
7.200E-05	\$	GEP248-M	\$	13	49	952
7.200E-05	\$	GEP2721M	\$	13	49	953
4.800E-06	\$	GFC234-M	\$	13	7	954
4.800E-06	\$	GFC236-M	\$	13	7	955
3.000E-04	\$	GFM001-F	\$	27	25	956
3.000E-04	\$	GFM002-F	\$	27	25	957
3.000E-04	\$	GFM003-F	\$	27	25	958
3.000E-04	\$	GFM004-F	\$	27	25	959

3.000E-04	\$	GFM005-F	\$	27	25	960
7.680E-07	\$	GFU-110H	\$	32	5	961
7.680E-07	\$	GFU-232H	\$	32	5	962
7.680E-07	\$	GFUACCRH	\$	32	5	963
7.680E-07	\$	GFUACEAH	\$	32	5	964
7.680E-07	\$	GFUMAXAH	\$	32	5	965
7.680E-07	\$	GFUMAXBH	\$	32	5	966
2.400E-05	\$	GIVSTATH	\$	6	5	967
7.200E-05	\$	GLX202-M	\$	13	13	968
7.200E-05	\$	GLX203-M	\$	13	13	969
7.200E-05	\$	GLX272-M	\$	13	13	970
2.496E-04	\$	GMG2021C	\$	18	14	971
2.496E-04	\$	GMG2022C	\$	18	14	972
2.496E-04	\$	GMG2023C	\$	18	14	973
2.496E-04	\$	GMG2024C	\$	18	14	974
2.496E-04	\$	GMG2025C	\$	18	14	975
2.496E-04	\$	GMG2031C	\$	18	14	976
2.496E-04	\$	GMG2032C	\$	18	14	977
2.496E-04	\$	GMG2033C	\$	18	14	978
2.496E-04	\$	GMG2034C	\$	18	14	979
2.496E-04	\$	GMG2035C	\$	18	14	980
1.200E-05	\$	GMG2041D	\$	18	14	981
1.200E-05	\$	GMG2042D	\$	18	14	982
1.200E-05	\$	GMG2043D	\$	18	14	983
1.200E-05	\$	GMG2044D	\$	18	14	984
1.200E-05	\$	GMG2045D	\$	18	14	985
2.496E-04	\$	GMG231AC	\$	18	14	986
2.496E-04	\$	GMG231BC	\$	18	14	987
2.496E-04	\$	GMG232AC	\$	18	14	988
2.496E-04	\$	GMG232BC	\$	18	14	989
2.496E-04	\$	GMG233AC	\$	18	14	990
2.496E-04	\$	GMG233BC	\$	18	14	991
2.496E-04	\$	GMG234AC	\$	18	14	992
2.496E-04	\$	GMG234BC	\$	18	14	993
2.496E-04	\$	GMG235AC	\$	18	14	994
2.496E-04	\$	GMG235BC	\$	18	14	995
2.496E-04	\$	GMT2051C	\$	48	18	996
2.496E-04	\$	GMT2052C	\$	48	18	997
3.000E-04	\$	GPM001-F	\$	27	25	998
1.200E-04	\$	GPM001-G	\$	32	25	999
3.000E-04	\$	GPM002-F	\$	27	25	1000
1.200E-04	\$	GPM002-G	\$	32	25	1001
3.000E-04	\$	GPM003-F	\$	27	25	1002
1.200E-04	\$	GPM003-G	\$	32	25	1003
3.000E-04	\$	GPM004-F	\$	27	25	1004
1.200E-04	\$	GPM004-G	\$	32	25	1005
3.000E-04	\$	GPM005-F	\$	27	25	1006
1.200E-04	\$	GPM005-G	\$	32	25	1007
8.160E-07	\$	GPP24HDL	\$	32	18	1008
8.160E-07	\$	GPP26HDL	\$	32	18	1009
1.000E-04	\$	GPS061-B	\$	13	14	1010
1.000E-04	\$	GPS062-B	\$	13	14	1011
1.000E-04	\$	GPS063-B	\$	13	14	1012
1.000E-04	\$	GPS064-B	\$	13	14	1013
1.000E-04	\$	GPS065-B	\$	13	14	1014
2.500E-04	\$	GPU001-G	\$	28	18	1015
2.500E-04	\$	GPU002-G	\$	28	18	1016
2.500E-04	\$	GPU003-G	\$	28	18	1017
2.500E-04	\$	GPU004-G	\$	28	18	1018
2.500E-04	\$	GPU005-G	\$	28	18	1019

1.128E-04	\$	GREBAILP	\$	4	9	1020
2.400E-06	\$	GRL2741H	\$	2	26	1021
2.400E-06	\$	GRL2742H	\$	2	26	1022
2.400E-06	\$	GRL2743H	\$	2	26	1023
2.400E-06	\$	GRL2744H	\$	2	26	1024
2.400E-06	\$	GRL2745H	\$	2	26	1025
2.400E-06	\$	GRL27X1H	\$	2	26	1026
2.400E-06	\$	GRL27X2H	\$	2	26	1027
2.400E-06	\$	GRL27X3H	\$	2	26	1028
2.400E-06	\$	GRL27X4H	\$	2	26	1029
2.400E-06	\$	GRL27X5H	\$	2	26	1030
7.200E-05	\$	GRL379-M	\$	2	53	1031
7.200E-05	\$	GRL380-M	\$	2	53	1032
7.200E-05	\$	GRL384-M	\$	2	53	1033
7.200E-05	\$	GRL384BM	\$	2	53	1034
2.400E-06	\$	GRL8009H	\$	2	26	1035
2.400E-06	\$	GRL8010H	\$	2	26	1036
2.400E-06	\$	GRLTT11H	\$	2	26	1037
2.400E-06	\$	GRLTT12H	\$	2	26	1038
2.400E-06	\$	GRLTT13H	\$	2	26	1039
2.400E-06	\$	GRLTT14H	\$	2	26	1040
2.400E-06	\$	GRLTT15H	\$	2	26	1041
2.400E-07	\$	GSG1A--L	\$	29	21	1042
2.400E-07	\$	GSG1B--L	\$	29	21	1043
2.400E-07	\$	GSG2A--L	\$	29	21	1044
2.400E-07	\$	GSG2B--L	\$	29	21	1045
2.400E-07	\$	GSG3A--L	\$	29	21	1046
2.400E-07	\$	GSG3B--L	\$	29	21	1047
2.400E-07	\$	GSG4A--L	\$	29	21	1048
2.400E-07	\$	GSG4B--L	\$	29	21	1049
2.400E-07	\$	GSG5A--L	\$	29	21	1050
2.400E-07	\$	GSG5B--L	\$	29	21	1051
1.000E-03	\$	GSV2481A	\$	13	25	1052
1.000E-03	\$	GSV2511A	\$	13	25	1053
1.000E-03	\$	GSV2512A	\$	13	25	1054
1.000E-03	\$	GSVTT11B	\$	20	25	1055
1.000E-03	\$	GSVTT12B	\$	20	25	1056
1.000E-03	\$	GSVTT21B	\$	20	25	1057
1.000E-03	\$	GSVTT22B	\$	20	25	1058
1.000E-03	\$	GSVTT31B	\$	20	25	1059
1.000E-03	\$	GSVTT32B	\$	20	25	1060
1.000E-03	\$	GSVTT41B	\$	20	25	1061
1.000E-03	\$	GSVTT42B	\$	20	25	1062
1.000E-03	\$	GSVTT51B	\$	20	25	1063
1.000E-03	\$	GSVTT52B	\$	20	25	1064
7.200E-05	\$	GSW204-H	\$	41	7	1065
7.200E-04	\$	GSW241-J	\$	41	7	1066
1.000E-02	\$	GSW241-Q	\$	41	7	1067
7.200E-04	\$	GSW245-J	\$	41	7	1068
1.000E-02	\$	GSW245-Q	\$	41	7	1069
1.000E-02	\$	GSW2H33Q	\$	41	7	1070
1.000E-04	\$	GSW3311B	\$	41	7	1071
1.000E-04	\$	GSW3312B	\$	41	7	1072
1.000E-04	\$	GSW3313B	\$	41	7	1073
1.000E-04	\$	GSW3314B	\$	41	7	1074
1.000E-04	\$	GSW3315B	\$	41	7	1075
1.000E-04	\$	GSW3321B	\$	41	7	1076
1.000E-04	\$	GSW3322B	\$	41	7	1077
1.000E-04	\$	GSW3323B	\$	41	7	1078
1.000E-04	\$	GSW3324B	\$	41	7	1079

1.000E-04	\$	GSW3325B	\$	41	7	1080
7.200E-05	\$	GSW5K65H	\$	41	5	1081
7.200E-04	\$	GSW749AJ	\$	41	7	1082
1.000E-04	\$	GSW749AV	\$	41	7	1083
7.200E-04	\$	GSW8009I	\$	41	7	1084
7.200E-04	\$	GSW8010I	\$	41	7	1085
1.000E-04	\$	GSWAUTOH	\$	41	5	1086
1.000E-04	\$	GSWTRANH	\$	41	5	1087
7.200E-05	\$	GTE248-M	\$	13	48	1088
7.200E-05	\$	GTE250-M	\$	13	48	1089
2.740E-10	\$	GTKPRESL	\$	29	48	1090
2.400E-05	\$	GXR1065P	\$	7	5	1091
2.400E-05	\$	GXR1165P	\$	7	5	1092
2.400E-05	\$	GXR416AP	\$	7	5	1093
2.400E-05	\$	GXR416BP	\$	7	5	1094
2.400E-05	\$	GXR7X65P	\$	7	5	1095
7.200E-05	\$	GZZTGN1P	\$	13	25	1096
7.200E-05	\$	GZZTGN2P	\$	13	25	1097
7.200E-05	\$	GZZTGN3P	\$	13	25	1098
7.200E-05	\$	GZZTGN4P	\$	13	25	1099
7.200E-05	\$	GZZTGN5P	\$	13	25	1100
1.205E-04	\$	HAV1051C	\$	20	50	1101
1.205E-04	\$	HAV1141C	\$	19	50	1102
1.205E-04	\$	HAV1241C	\$	19	46	1103
1.205E-04	\$	HAV1242C	\$	19	46	1104
1.205E-04	\$	HAVCV01E	\$	19	46	1105
1.205E-04	\$	HAVCV02E	\$	19	46	1106
8.568E-04	\$	HBRCE-1P	\$	29	4	1107
1.918E-05	\$	HBRCE-1X	\$	32	4	1108
8.568E-04	\$	HBRCE-2P	\$	29	4	1109
1.918E-05	\$	HBRCE-2X	\$	32	4	1110
8.568E-04	\$	HBRFW-1P	\$	29	4	1111
1.918E-05	\$	HBRFW-1X	\$	32	4	1112
7.200E-06	\$	HBU138AH	\$	46	32	1113
7.200E-06	\$	HBU138BH	\$	46	32	1114
7.200E-06	\$	HBU416AH	\$	47	32	1115
7.200E-06	\$	HBU416BH	\$	47	32	1116
7.200E-06	\$	HBU480AH	\$	47	32	1117
7.200E-06	\$	HBU480BH	\$	47	32	1118
7.200E-05	\$	HCLAIR-L	\$	23	46	1119
7.200E-05	\$	HCLOILSL	\$	24	46	1120
7.200E-04	\$	HCLOILTE	\$	24	46	1121
7.200E-08	\$	HCLOILTL	\$	24	46	1122
7.200E-08	\$	HCLTUBEL	\$	24	46	1123
7.200E-05	\$	HCMTGHWP	\$	13	46	1124
7.200E-04	\$	HCPP004G	\$	32	52	1125
5.280E-05	\$	HCV1041C	\$	21	46	1126
5.280E-05	\$	HCV1042C	\$	21	46	1127
3.300E-05	\$	HCV1081A	\$	22	46	1128
3.300E-05	\$	HCV1082A	\$	22	46	1129
5.280E-05	\$	HCV1221C	\$	21	46	1130
5.280E-05	\$	HCV1222C	\$	21	46	1131
5.280E-05	\$	HCV1541C	\$	21	46	1132
5.280E-05	\$	HCV1551C	\$	21	46	1133
3.300E-05	\$	HCVCV04A	\$	22	46	1134
5.280E-05	\$	HCVCV05C	\$	22	50	1135
3.300E-05	\$	HCVCV06A	\$	22	46	1136
3.300E-05	\$	HCVCV07A	\$	22	46	1137
2.359E-05	\$	HDV1051C	\$	20	46	1138
2.472E-05	\$	HFV0101C	\$	21	45	1139

1.286E-05	\$ HGV1072C	\$	22	46	1140
1.286E-05	\$ HGV1073C	\$	22	46	1141
1.286E-05	\$ HGV1119C	\$	22	46	1142
1.286E-05	\$ HGV1120C	\$	22	46	1143
7.512E-06	\$ HMB0102C	\$	18	46	1144
2.496E-04	\$ HMG1071C	\$	18	50	1145
2.496E-04	\$ HMG1251C	\$	19	46	1146
2.496E-04	\$ HMG1252C	\$	19	46	1147
2.496E-04	\$ HMG1431C	\$	18	46	1148
2.496E-04	\$ HMG1432C	\$	18	46	1149
5.000E-04	\$ HMP138BF	\$	16	47	1150
7.200E-04	\$ HMP138BG	\$	32	47	1151
1.918E-05	\$ HMP138BX	\$	32	47	1152
7.200E-04	\$ HMPP003G	\$	32	47	1153
5.000E-04	\$ HMPP005F	\$	16	47	1154
7.200E-04	\$ HMPP005G	\$	32	47	1155
1.918E-05	\$ HMPP005X	\$	32	47	1156
8.352E-05	\$ HMPTGCAG	\$	32	47	1157
8.352E-05	\$ HMPTGCBG	\$	32	47	1158
7.200E-04	\$ HMPVP01G	\$	32	47	1159
5.000E-04	\$ HMPVP02F	\$	16	47	1160
7.200E-04	\$ HMPVP02G	\$	32	47	1161
1.918E-05	\$ HMPVP02X	\$	32	47	1162
2.496E-04	\$ HMPV1051C	\$	18	51	1163
1.205E-04	\$ HPV1071C	\$	20	50	1164
1.205E-04	\$ HPV1081C	\$	20	50	1165
2.370E-04	\$ HTG138BG	\$	32	52	1166
7.820E-05	\$ HTG138BP	\$	10	52	1167
1.286E-05	\$ HTV1011C	\$	21	46	1168
1.286E-05	\$ HTV1012C	\$	21	46	1169
1.286E-05	\$ HTV1031C	\$	21	50	1170
1.286E-05	\$ HTV1041C	\$	21	46	1171
1.286E-05	\$ HTV1042C	\$	21	46	1172
1.286E-05	\$ HTV1043C	\$	21	46	1173
1.286E-05	\$ HTV1044C	\$	21	46	1174
1.286E-05	\$ HTV1201C	\$	21	46	1175
1.286E-05	\$ HTV1501C	\$	21	46	1176
1.286E-05	\$ HTV1521C	\$	21	46	1177
1.286E-05	\$ HTV1531C	\$	21	46	1178
1.286E-05	\$ HTV2072C	\$	21	28	1179
1.286E-05	\$ HTV2073C	\$	21	28	1180
2.370E-04	\$ HZZBUSAP	\$	1	1	1181
8.820E-03	\$ IE-10	\$	32	1	1182
2.660E-03	\$ IE-11	\$	32	1	1183
8.700E-02	\$ IE-12A	\$	32	1	1184
7.500E-03	\$ IE-12B	\$	32	1	1185
8.700E-07	\$ IE-13A	\$	32	1	1186
3.500E-02	\$ IE-13B	\$	32	1	1187
3.500E-02	\$ IE-13C	\$	32	1	1188
1.700E-02	\$ IE-13D-1	\$	32	1	1189
1.700E-02	\$ IE-13D-2	\$	32	1	1190
1.700E-02	\$ IE-13D-3	\$	32	1	1191
1.700E-02	\$ IE-13D-5	\$	32	1	1192
1.700E-02	\$ IE-13D-6	\$	32	1	1193
4.300E-03	\$ IE-13E-1	\$	32	1	1194
4.300E-03	\$ IE-13E-2	\$	32	1	1195
4.300E-03	\$ IE-13E-3	\$	32	1	1196
1.000E+00	\$ IE-1A	\$	32	1	1197
6.100E-01	\$ IE-1B	\$	32	1	1198
9.400E-04	\$ IE-2	\$	32	1	1199

3.020E-06	\$ IE-5	\$	32	1	1200
1.190E-05	\$ IE-6	\$	32	1	1201
1.700E-01	\$ IE-7A	\$	32	1	1202
8.700E-02	\$ IE-7B	\$	32	1	1203
5.800E-07	\$ IE-8A	\$	32	1	1204
4.300E-02	\$ IE-8B	\$	32	1	1205
3.000E-01	\$ IE-9	\$	32	1	1206
1.000E-09	\$ INTFLOCA	\$	32	1	1207
1.128E-04	\$ JBC105AH	\$	4	7	1208
1.128E-04	\$ JBC105BH	\$	4	7	1209
1.128E-04	\$ JBC105DH	\$	4	7	1210
1.128E-04	\$ JBC109AH	\$	4	5	1211
1.128E-04	\$ JBC109BH	\$	4	25	1212
1.128E-04	\$ JBC153AH	\$	4	31	1213
1.128E-04	\$ JBC153BH	\$	4	31	1214
3.120E-04	\$ JBT105AK	\$	4	7	1215
3.120E-04	\$ JBT105BK	\$	4	7	1216
3.120E-04	\$ JBT105DK	\$	4	7	1217
3.120E-04	\$ JBT105EK	\$	4	7	1218
3.120E-04	\$ JBT109AK	\$	4	25	1219
3.120E-04	\$ JBT109BK	\$	4	25	1220
3.120E-04	\$ JBT153AK	\$	4	31	1221
3.120E-04	\$ JBT153BK	\$	4	31	1222
7.200E-06	\$ JBU105AH	\$	12	8	1223
7.200E-06	\$ JBU105BH	\$	12	8	1224
7.200E-06	\$ JBU105DH	\$	12	8	1225
7.200E-06	\$ JBU105EH	\$	12	8	1226
7.200E-06	\$ JBU109AH	\$	12	25	1227
7.200E-06	\$ JBU109BH	\$	12	25	1228
7.200E-06	\$ JBU153AH	\$	12	30	1229
7.200E-06	\$ JBU153BH	\$	12	30	1230
1.000E-04	\$ JSW109AH	\$	12	25	1231
1.000E-04	\$ JSW109BH	\$	12	25	1232
3.000E-03	\$ KBCRW3K	\$	3	32	1233
4.800E-04	\$ KCLCRM1P	\$	32	40	1234
4.800E-04	\$ KCLCRM2P	\$	32	40	1235
4.800E-04	\$ KCLCRM3P	\$	32	40	1236
4.800E-04	\$ KCLCRM4P	\$	32	40	1237
1.100E-03	\$ KCV1061A	\$	21	46	1238
1.100E-03	\$ KCV1062A	\$	21	46	1239
1.100E-03	\$ KCV5031A	\$	21	40	1240
1.100E-03	\$ KCV5032A	\$	21	40	1241
1.704E-05	\$ KCV5051C	\$	21	40	1242
1.704E-05	\$ KCV5052C	\$	21	40	1243
1.100E-03	\$ KCV5053A	\$	21	40	1244
1.704E-05	\$ KCV5054C	\$	21	40	1245
1.100E-03	\$ KCV5081A	\$	21	40	1246
1.100E-03	\$ KCV5082A	\$	21	40	1247
1.704E-05	\$ KCV5511C	\$	21	31	1248
1.100E-03	\$ KCV5531A	\$	21	31	1249
1.100E-03	\$ KCV5533A	\$	21	29	1250
1.100E-03	\$ KCV553FA	\$	21	31	1251
1.100E-03	\$ KCV5601A	\$	21	29	1252
1.100E-03	\$ KCV5602A	\$	21	29	1253
1.704E-05	\$ KCV5651C	\$	21	31	1254
1.704E-05	\$ KCV5652C	\$	21	31	1255
1.100E-03	\$ KCV5653A	\$	21	31	1256
1.704E-05	\$ KCV5653C	\$	21	31	1257
1.704E-05	\$ KCV5691C	\$	21	31	1258
1.704E-05	\$ KCV5692C	\$	21	31	1259

1.100E-03	\$	KCV5693A	\$	21	31	1260
1.704E-05	\$	KCV5693C	\$	21	31	1261
1.704E-05	\$	KCV5734C	\$	19	31	1262
1.704E-05	\$	KCV5735C	\$	19	31	1263
1.100E-03	\$	KCV5861A	\$	21	31	1264
1.100E-03	\$	KCV5950A	\$	19	31	1265
1.704E-05	\$	KCV6184C	\$	21	29	1266
1.704E-05	\$	KCV6185C	\$	21	29	1267
1.704E-05	\$	KCV8061C	\$	21	40	1268
1.704E-05	\$	KCV8062C	\$	21	40	1269
1.100E-03	\$	KCV8063A	\$	21	40	1270
1.704E-05	\$	KCV8063C	\$	21	40	1271
1.704E-05	\$	KCV8064C	\$	21	40	1272
4.500E-04	\$	KDPFSD1F	\$	16	29	1273
2.640E-02	\$	KDPFSD1G	\$	32	29	1274
4.500E-04	\$	KDPFSD2F	\$	16	29	1275
2.640E-02	\$	KDPFSD2G	\$	32	29	1276
8.160E-07	\$	KDUF1INE	\$	21	32	1277
8.160E-07	\$	KDUF1OUE	\$	21	32	1278
8.160E-07	\$	KDUF2INE	\$	21	32	1279
8.160E-07	\$	KDUF2OUE	\$	21	32	1280
6.240E-05	\$	KDV6044C	\$	20	29	1281
6.240E-05	\$	KDV6045C	\$	20	29	1282
2.640E-06	\$	KFV5522C	\$	21	31	1283
2.640E-06	\$	KFV5523C	\$	21	29	1284
2.640E-06	\$	KFV5524C	\$	21	29	1285
2.640E-06	\$	KFV5551C	\$	21	31	1286
1.100E-03	\$	KFV5552A	\$	21	31	1287
1.100E-03	\$	KFV5572A	\$	21	31	1288
2.640E-06	\$	KFV5611C	\$	21	29	1289
2.640E-06	\$	KFV5612C	\$	21	29	1290
2.640E-06	\$	KFV5631C	\$	21	31	1291
2.640E-06	\$	KFV5632C	\$	21	31	1292
2.640E-06	\$	KFV5633C	\$	21	31	1293
2.640E-06	\$	KFV5661C	\$	21	31	1294
2.640E-06	\$	KFV5662C	\$	21	31	1295
2.640E-06	\$	KFV5663C	\$	21	31	1296
2.640E-06	\$	KFV5671C	\$	21	31	1297
2.640E-06	\$	KFV5671C	\$	21	31	1298
2.640E-06	\$	KFV5672C	\$	21	31	1299
2.640E-06	\$	KFV5673C	\$	21	31	1300
2.640E-06	\$	KFV5674C	\$	21	31	1301
2.640E-06	\$	KFV5681C	\$	21	31	1302
2.640E-06	\$	KFV5682C	\$	21	31	1303
2.640E-06	\$	KFV5683C	\$	21	31	1304
2.640E-06	\$	KGTCRW2C	\$	20	34	1305
2.640E-06	\$	KGTCRW4C	\$	20	34	1306
2.640E-06	\$	KGTL1D1C	\$	20	34	1307
2.640E-06	\$	KGTL1D2C	\$	20	34	1308
2.640E-06	\$	KGTRGP1C	\$	21	34	1309
2.640E-06	\$	KGTRGP2C	\$	21	34	1310
2.640E-06	\$	KGTRGP3C	\$	21	34	1311
2.640E-06	\$	KGTRGP4C	\$	21	34	1312
2.640E-06	\$	KGTS GP1C	\$	21	34	1313
2.640E-06	\$	KGTS GP4C	\$	21	34	1314
2.640E-06	\$	KG V4601C	\$	22	29	1315
2.640E-06	\$	KG V5031C	\$	22	40	1316
2.640E-06	\$	KG V5032C	\$	22	40	1317
2.640E-06	\$	KG V5033C	\$	22	40	1318
2.640E-06	\$	KG V5034C	\$	22	40	1319

2.640E-06	\$	KGVS041C	\$	21	40	1320
2.640E-06	\$	KGVS042C	\$	21	40	1321
2.640E-06	\$	KGVS063C	\$	22	40	1322
2.640E-06	\$	KGVS601C	\$	22	29	1323
2.640E-06	\$	KGVS6004C	\$	21	29	1324
2.640E-06	\$	KGVS6005C	\$	21	29	1325
2.640E-06	\$	KGVS6014C	\$	22	29	1326
2.640E-06	\$	KGVS6015C	\$	22	29	1327
2.640E-06	\$	KGVS6164C	\$	20	29	1328
2.640E-06	\$	KGVS6165C	\$	20	29	1329
2.640E-06	\$	KGVS6174C	\$	22	29	1330
2.640E-06	\$	KGVS6175C	\$	22	29	1331
2.400E-04	\$	KHB8051C	\$	20	40	1332
2.400E-04	\$	KHB8052C	\$	20	40	1333
1.100E-02	\$	KHB8053A	\$	20	40	1334
2.400E-04	\$	KHB8053C	\$	20	40	1335
2.400E-04	\$	KHB8054C	\$	20	40	1336
1.000E-04	\$	KLS554XO	\$	19	29	1337
1.000E-04	\$	KLS555XO	\$	19	29	1338
2.040E-05	\$	KMB8011C	\$	20	1	1339
2.040E-05	\$	KMB8012C	\$	20	1	1340
2.040E-05	\$	KMB8021C	\$	20	1	1341
2.040E-05	\$	KMB8022C	\$	20	1	1342
6.600E-04	\$	KMPCRW1G	\$	17	40	1343
6.600E-04	\$	KMPCRW2G	\$	17	40	1344
6.800E-03	\$	KMPCRW3F	\$	17	40	1345
6.600E-04	\$	KMPCRW3G	\$	17	40	1346
1.920E-06	\$	KMPCRW3W	\$	32	40	1347
1.920E-06	\$	KMPCRW3X	\$	32	40	1348
6.600E-04	\$	KMPCRW4G	\$	17	40	1349
8.700E-04	\$	KMPHRP1G	\$	16	32	1350
8.700E-04	\$	KMPHRP2G	\$	16	32	1351
2.500E-03	\$	KMPHRP2F	\$	16	32	1352
8.700E-04	\$	KMPHRP3G	\$	16	32	1353
1.920E-06	\$	KMPHRP3W	\$	32	32	1354
1.920E-06	\$	KMPHRP3X	\$	32	32	1355
8.700E-04	\$	KMPLRP1G	\$	32	32	1356
8.700E-04	\$	KMPLRP2G	\$	32	32	1357
1.300E-03	\$	KMPLRP3F	\$	16	32	1358
8.700E-04	\$	KMPLRP3G	\$	32	32	1359
1.920E-06	\$	KMPLRP3W	\$	32	32	1360
1.920E-06	\$	KMPLRP3X	\$	32	32	1361
6.600E-04	\$	KMPSER1F	\$	16	47	1362
5.100E-03	\$	KMPSER1G	\$	32	47	1363
6.600E-04	\$	KMPSER2F	\$	16	47	1364
5.100E-03	\$	KMPSER2G	\$	32	47	1365
2.000E-03	\$	KMPSWP1F	\$	16	42	1366
8.300E-05	\$	KMPSWP1G	\$	32	42	1367
1.920E-06	\$	KMPSWP1X	\$	32	42	1368
2.000E-03	\$	KMPSWP2F	\$	16	42	1369
8.300E-05	\$	KMPSWP2G	\$	32	42	1370
1.920E-06	\$	KMPSWP2X	\$	32	42	1371
1.000E+00	\$	KOP5552R	\$	42	42	1372
1.000E+00	\$	KOP5572R	\$	42	42	1373
1.000E+00	\$	KOP5964R	\$	42	42	1374
1.000E+00	\$	KOP5965R	\$	42	42	1375
1.000E+00	\$	KOPCRW3R	\$	42	42	1376
1.000E-03	\$	KOPFSD1V	\$	32	1	1377
1.000E-03	\$	KOPFSD2V	\$	32	1	1378
1.000E+00	\$	KOPFSDPQ	\$	42	42	1379

1.000E+00	\$	KOPHRP3Q	\$	42	42	1380
1.000E+00	\$	KOPLRP3Q	\$	42	42	1381
1.000E+00	\$	KOPSBUSQ	\$	42	42	1382
1.000E+00	\$	KOPSERWQ	\$	42	42	1383
1.000E+00	\$	KOPSWPSQ	\$	42	42	1384
8.160E-07	\$	KPP1020E	\$	32	1	1385
8.160E-07	\$	KPP1080E	\$	32	1	1386
8.160E-07	\$	KPP6600E	\$	32	1	1387
8.160E-07	\$	KPPCRWSL	\$	32	1	1388
8.160E-07	\$	KPPEMERE	\$	32	1	1389
8.160E-07	\$	KPPFAB1L	\$	32	1	1390
8.160E-07	\$	KPPFAB2L	\$	32	1	1391
8.160E-07	\$	KPPFS1FL	\$	32	1	1392
8.160E-07	\$	KPPFS2FL	\$	32	1	1393
8.160E-07	\$	KPPFSA1L	\$	32	1	1394
8.160E-07	\$	KPPFSA2L	\$	32	1	1395
8.160E-07	\$	KPPFSDPL	\$	32	1	1396
8.160E-07	\$	KPPHCL1L	\$	32	1	1397
8.160E-07	\$	KPPHCL2L	\$	32	1	1398
8.160E-07	\$	KPPLRDHL	\$	32	1	1399
8.160E-07	\$	KPPRWS2L	\$	32	1	1400
8.160E-07	\$	KPPSCL1L	\$	32	1	1401
8.160E-07	\$	KPPSCL2L	\$	32	1	1402
8.160E-07	\$	KPPSRWRL	\$	32	1	1403
1.000E-04	\$	KPS116XM	\$	13	46	1404
1.000E-04	\$	KPS50040	\$	13	29	1405
1.000E-04	\$	KPS50050	\$	13	29	1406
1.000E-04	\$	KPS58400	\$	13	31	1407
1.000E-04	\$	KPS584AO	\$	13	31	1408
1.000E-04	\$	KPSCR10	\$	13	40	1409
1.000E-04	\$	KPSCR20	\$	13	40	1410
1.000E-04	\$	KPSCR30	\$	13	40	1411
1.000E-04	\$	KPSCR40	\$	13	40	1412
7.500E-04	\$	KPV6151A	\$	19	29	1413
2.500E-01	\$	KSCREEN	\$	42	1	1414
8.160E-07	\$	KSL0001E	\$	0	1	1415
8.160E-07	\$	KSL0002E	\$	0	1	1416
4.080E-05	\$	KSV5624C	\$	31	29	1417
4.080E-05	\$	KSV5625C	\$	31	29	1418
6.200E-04	\$	KSV6004A	\$	31	29	1419
6.200E-04	\$	KSV6005A	\$	31	29	1420
6.200E-04	\$	KSVFS1SA	\$	19	31	1421
6.200E-04	\$	KSVFS2SA	\$	19	31	1422
1.000E-04	\$	KSW116XO	\$	41	46	1423
1.000E-04	\$	KSWSWP10	\$	41	40	1424
1.000E-04	\$	KSWSWP20	\$	41	40	1425
2.040E-05	\$	KTCKPOF1L	\$	23	1	1426
2.040E-05	\$	KTCKPOF2L	\$	23	1	1427
2.040E-05	\$	KTCKFS1AL	\$	30	31	1428
2.040E-05	\$	KTCKFS1XL	\$	30	29	1429
2.040E-05	\$	KTCKFS2AL	\$	30	31	1430
2.040E-05	\$	KTCKFS2XL	\$	23	29	1431
2.040E-05	\$	KTCKFSABL	\$	5	31	1432
3.000E-04	\$	KTNCRP1F	\$	33	42	1433
1.200E-04	\$	KTNCRP1G	\$	33	42	1434
3.000E-04	\$	KTNCRP2F	\$	33	42	1435
1.200E-04	\$	KTNCRP2G	\$	33	42	1436
3.000E-04	\$	KTNCRP3F	\$	33	42	1437
1.200E-04	\$	KTNCRP3G	\$	33	42	1438
3.000E-04	\$	KTNCRP4F	\$	33	42	1439

1.200E-04	\$	KTNCRP4G	\$	33	42	1440
3.000E-04	\$	KT5524A	\$	22	29	1441
3.000E-04	\$	KT5525A	\$	22	29	1442
2.640E-06	\$	KTV1021C	\$	21	46	1443
2.640E-06	\$	KTV1051C	\$	21	46	1444
2.640E-06	\$	KTV1052C	\$	21	46	1445
2.640E-06	\$	KTV1053C	\$	21	46	1446
2.640E-06	\$	KTV1054C	\$	21	46	1447
2.640E-06	\$	KTV5021C	\$	21	40	1448
2.640E-06	\$	KTV5022C	\$	21	40	1449
2.640E-06	\$	KTV5081C	\$	22	40	1450
2.640E-06	\$	KTV5511C	\$	22	31	1451
2.640E-06	\$	KTV5511C	\$	22	31	1452
2.640E-06	\$	KTV5512C	\$	22	31	1453
2.640E-06	\$	KTV5512C	\$	22	31	1454
2.640E-06	\$	KTV5531C	\$	22	31	1455
2.640E-06	\$	KTV5532C	\$	22	31	1456
2.640E-06	\$	KTV5533C	\$	22	31	1457
2.640E-06	\$	KTV5534C	\$	21	31	1458
2.640E-06	\$	KTV5581C	\$	22	31	1459
2.640E-06	\$	KTV5582C	\$	22	31	1460
2.640E-06	\$	KTV5964C	\$	22	31	1461
2.640E-06	\$	KTV5965C	\$	22	31	1462
2.640E-06	\$	KTV599XC	\$	21	31	1463
2.640E-06	\$	KTV6231C	\$	21	29	1464
2.640E-06	\$	KTV8031C	\$	22	31	1465
2.640E-06	\$	KTV8031C	\$	31	31	1466
2.640E-06	\$	KTV8164C	\$	22	31	1467
2.640E-06	\$	KTV8165C	\$	31	31	1468
2.640E-06	\$	KTV8172C	\$	22	31	1469
2.640E-06	\$	KTV8172C	\$	22	31	1470
2.400E-05	\$	KZZHYD2P	\$	16	40	1471
4.800E-04	\$	LCLHFP1P	\$	32	1	1472
4.800E-04	\$	LCLHFP2P	\$	32	1	1473
4.800E-04	\$	LCLHFP3P	\$	32	1	1474
4.800E-04	\$	LCLLFP1P	\$	32	1	1475
4.800E-04	\$	LCLLFP2P	\$	32	1	1476
4.800E-04	\$	LCLLFP3P	\$	32	1	1477
4.800E-04	\$	LCLLFP4P	\$	32	1	1478
2.040E-05	\$	LCV5571C	\$	21	29	1479
2.040E-05	\$	LCV5572C	\$	21	29	1480
2.040E-05	\$	LCV5573C	\$	21	29	1481
2.040E-05	\$	LCV5574C	\$	21	29	1482
1.200E-05	\$	LCV5581D	\$	21	31	1483
3.000E-04	\$	LCV5591C	\$	21	31	1484
3.000E-04	\$	LCV5592C	\$	21	31	1485
3.000E-04	\$	LCV5593C	\$	21	31	1486
7.200E-06	\$	LFV5511C	\$	21	31	1487
7.200E-06	\$	LFV5512C	\$	21	29	1488
7.200E-06	\$	LFV5521C	\$	21	29	1489
7.200E-06	\$	LFV5522C	\$	21	29	1490
7.200E-06	\$	LFV5523C	\$	21	29	1491
7.200E-06	\$	LFV5524C	\$	21	29	1492
7.200E-06	\$	LFV5541C	\$	21	29	1493
7.200E-06	\$	LFV5542C	\$	21	29	1494
7.200E-06	\$	LFV5543C	\$	21	29	1495
7.200E-06	\$	LFV5544C	\$	21	29	1496
7.200E-06	\$	LFV5551C	\$	21	31	1497
7.200E-06	\$	LFV5552C	\$	21	31	1498
7.200E-06	\$	LFV5553C	\$	21	31	1499

7.200E-06	\$	LFV8112C	\$	21	1	1500
1.000E+00	\$	LLOCA	\$	32	1	1501
5.100E-05	\$	LMPHFP1G	\$	32	32	1502
5.100E-05	\$	LMPHFP2G	\$	32	32	1503
3.000E-03	\$	LMPHFP3F	\$	16	32	1504
5.100E-05	\$	LMPHFP3G	\$	32	32	1505
1.918E-05	\$	LMPHFP3W	\$	32	32	1506
1.918E-05	\$	LMPHFP3X	\$	32	32	1507
2.500E-04	\$	LMPLFP1G	\$	32	30	1508
2.500E-04	\$	LMPLFP2G	\$	32	30	1509
8.300E-03	\$	LMPLFP3F	\$	16	30	1510
2.500E-04	\$	LMPLFP3G	\$	32	30	1511
1.918E-05	\$	LMPLFP3W	\$	32	30	1512
1.918E-05	\$	LMPLFP3X	\$	32	30	1513
8.300E-03	\$	LMPLFP4F	\$	16	30	1514
8.300E-03	\$	LMPLFP4F	\$	16	30	1515
2.500E-04	\$	LMPLFP4G	\$	32	30	1516
1.918E-05	\$	LMPLFP4W	\$	32	30	1517
1.918E-05	\$	LMPLFP4X	\$	32	30	1518
0.400E+00	\$	LOPHFP3Q	\$	42	32	1519
0.400E+00	\$	LOPLFP3R	\$	42	30	1520
0.400E+00	\$	LOPLFP4R	\$	42	30	1521
1.000E+00	\$	LOPSWITQ	\$	42	30	1522
8.160E-07	\$	LPPFWS1L	\$	32	1	1523
2.040E-05	\$	LTKFWT0L	\$	26	1	1524
1.000E-04	\$	LTV5561C	\$	21	31	1525
1.000E-04	\$	LTV5562C	\$	21	31	1526
1.000E-04	\$	LTV5563C	\$	21	31	1527
1.200E-05	\$	LTV5591D	\$	21	29	1528
4.800E-04	\$	MCLAHR1P	\$	32	1	1529
4.800E-04	\$	MCLAHR2P	\$	32	1	1530
4.800E-04	\$	MCLAHR3P	\$	32	1	1531
4.800E-04	\$	MCLSLM1P	\$	32	1	1532
4.800E-04	\$	MCLSLM2P	\$	32	1	1533
5.500E-05	\$	MCV5521C	\$	21	31	1534
5.500E-05	\$	MCV5522C	\$	21	31	1535
5.500E-05	\$	MCV5523C	\$	21	31	1536
1.368E-05	\$	MCV5531C	\$	22	31	1537
1.368E-05	\$	MCV5532C	\$	22	31	1538
1.368E-05	\$	MCV5571C	\$	21	31	1539
1.368E-05	\$	MCV5572C	\$	21	31	1540
1.368E-05	\$	MCV5573C	\$	21	31	1541
1.368E-05	\$	MCV5641C	\$	21	31	1542
8.200E-06	\$	MDV5621C	\$	20	31	1543
8.200E-06	\$	MDV562XC	\$	20	31	1544
2.000E-05	\$	MFV5511C	\$	21	31	1545
2.000E-05	\$	MFV5512C	\$	21	31	1546
2.000E-05	\$	MFV5513C	\$	21	31	1547
2.000E-05	\$	MFV5521C	\$	21	31	1548
4.800E-04	\$	MFV5522A	\$	21	31	1549
2.000E-05	\$	MFV5523C	\$	21	31	1550
4.800E-04	\$	MFV5524A	\$	21	31	1551
2.000E-05	\$	MFV5526C	\$	21	31	1552
2.640E-05	\$	MFV8041C	\$	21	1	1553
2.640E-05	\$	MFV8053C	\$	21	1	1554
2.640E-05	\$	MFV8131C	\$	21	1	1555
1.000E-04	\$	MLSSTLVO	\$	13	32	1556
6.100E-03	\$	MMPAHR1F	\$	16	32	1557
9.000E-05	\$	MMPAHR1G	\$	32	32	1558
1.918E-05	\$	MMPAHR1W	\$	32	32	1559

1.918E-05	\$	MMPAHR1X	\$	32	32	1560
6.100E-03	\$	MMPAHR2F	\$	16	32	1561
9.000E-05	\$	MMPAHR2G	\$	32	32	1562
1.918E-05	\$	MMPAHR2W	\$	32	32	1563
1.918E-05	\$	MMPAHR2X	\$	32	32	1564
6.100E-03	\$	MMPAHR3F	\$	16	32	1565
9.000E-05	\$	MMPAHR3G	\$	32	32	1566
1.918E-05	\$	MMPAHR3W	\$	32	32	1567
1.918E-05	\$	MMPAHR3X	\$	32	32	1568
7.200E-04	\$	MMPLPI1G	\$	16	32	1569
7.200E-04	\$	MMPLPI2G	\$	16	32	1570
3.000E-03	\$	MMPLPI3F	\$	16	32	1571
7.200E-04	\$	MMPLPI3G	\$	32	32	1572
1.918E-05	\$	MMPLPI3W	\$	32	32	1573
1.918E-05	\$	MMPLPI3X	\$	32	32	1574
8.300E-05	\$	MMPSLM1G	\$	16	32	1575
8.300E-05	\$	MMPSLM2G	\$	16	32	1576
1.000E+00	\$	MOP5522Q	\$	42	1	1577
1.000E+00	\$	MOP5524Q	\$	42	1	1578
1.000E-03	\$	MOPAHDV	\$	32	1	1579
1.000E+00	\$	MOPAHRPR	\$	42	1	1580
1.000E+00	\$	MOPLBUSQ	\$	42	1	1581
1.000E+00	\$	MOPLPI3Q	\$	42	1	1582
1.000E+00	\$	MOPSBUSQ	\$	42	1	1583
8.160E-07	\$	MPPAHRSL	\$	32	1	1584
8.160E-07	\$	MPPLPISL	\$	32	1	1585
8.160E-07	\$	MPPSLMUL	\$	32	1	1586
2.040E-05	\$	MTKAHRTL	\$	26	1	1587
2.040E-05	\$	MTKAHRTL	\$	26	1	1588
1.700E-06	\$	MTKDWTOL	\$	26	1	1589
1.700E-06	\$	MTKDWTOL	\$	26	1	1590
1.700E-06	\$	MTKDWTOL	\$	26	1	1591
8.160E-06	\$	MTV5541C	\$	21	31	1592
8.160E-06	\$	MTV5542C	\$	21	31	1593
8.160E-06	\$	MTV5551C	\$	21	31	1594
8.160E-06	\$	MTV5552C	\$	21	31	1595
8.160E-06	\$	MTV5553C	\$	21	31	1596
8.160E-06	\$	MTV5554C	\$	21	31	1597
8.160E-06	\$	MTV5555C	\$	21	31	1598
8.160E-06	\$	MTV5561C	\$	21	31	1599
8.160E-06	\$	MTV5562C	\$	21	31	1600
8.160E-06	\$	MTV5563C	\$	21	31	1601
5.600E-04	\$	NAG0185A	\$	20	46	1602
6.240E-05	\$	NAG1031E	\$	20	46	1603
5.600E-04	\$	NAG1061A	\$	20	46	1604
5.600E-04	\$	NAG1081A	\$	20	46	1605
7.200E-04	\$	NCRA1CLP	\$	32	47	1606
7.200E-04	\$	NCRA2CLP	\$	32	47	1607
3.500E-03	\$	NCRCOM1F	\$	16	47	1608
6.000E-04	\$	NCRCOM1G	\$	32	47	1609
3.500E-03	\$	NCRCOM2F	\$	16	47	1610
6.000E-04	\$	NCRCOM2G	\$	32	47	1611
6.000E-04	\$	NCRIA11G	\$	32	47	1612
6.000E-04	\$	NCRIA12G	\$	32	47	1613
6.000E-04	\$	NCRSAIRG	\$	32	47	1614
1.704E-05	\$	NCV0104E	\$	21	46	1615
1.704E-05	\$	NCV1011E	\$	21	46	1616
1.704E-05	\$	NCV1013E	\$	21	46	1617
1.704E-05	\$	NCV1071C	\$	21	46	1618
1.704E-05	\$	NCV1221E	\$	21	46	1619

1.704E-05	\$	NCV1222E	\$	21	46	1620
1.704E-05	\$	NCV1901C	\$	21	46	1621
1.704E-05	\$	NCVCVA1E	\$	22	7	1622
1.704E-05	\$	NCVCVB1E	\$	22	7	1623
7.200E-04	\$	NDYCOM1E	\$	32	1	1624
7.200E-04	\$	NDYCOM2E	\$	32	1	1625
7.200E-04	\$	NDYDOT1E	\$	22	7	1626
7.200E-04	\$	NDYDOT2E	\$	22	7	1627
7.200E-04	\$	NDYIAIRE	\$	30	46	1628
7.200E-04	\$	NFLAD02E	\$	30	46	1629
7.200E-04	\$	NFLCOM1E	\$	32	1	1630
7.200E-04	\$	NFLCOM2E	\$	32	1	1631
7.200E-04	\$	NFLIAC1E	\$	32	46	1632
7.200E-04	\$	NFLIAC2E	\$	32	46	1633
7.200E-04	\$	NFLTR11E	\$	30	46	1634
8.160E-05	\$	NGV1001E	\$	21	46	1635
8.160E-05	\$	NGV1003E	\$	21	46	1636
8.160E-05	\$	NGV1081C	\$	22	46	1637
8.160E-05	\$	NGV1082C	\$	22	46	1638
8.160E-05	\$	NGV1083C	\$	22	46	1639
8.160E-05	\$	NGV1084C	\$	22	46	1640
8.160E-05	\$	NGV1087C	\$	22	46	1641
8.160E-05	\$	NGV1088C	\$	22	46	1642
1.200E-05	\$	NGV1093D	\$	22	46	1643
8.160E-05	\$	NGV1153E	\$	22	46	1644
8.160E-05	\$	NGV1154E	\$	22	46	1645
8.160E-05	\$	NGV1156C	\$	22	46	1646
8.160E-05	\$	NGV1158C	\$	22	46	1647
8.160E-05	\$	NGV1159C	\$	22	46	1648
8.160E-05	\$	NGV1413E	\$	22	46	1649
8.160E-05	\$	NGV1414E	\$	22	46	1650
8.160E-05	\$	NGV1511E	\$	22	46	1651
8.160E-05	\$	NGV1512C	\$	22	46	1652
8.160E-05	\$	NGV1513E	\$	22	46	1653
8.160E-05	\$	NGV1514E	\$	22	46	1654
8.160E-05	\$	NGV1515E	\$	22	46	1655
8.160E-05	\$	NGV1518C	\$	22	46	1656
8.160E-05	\$	NGV1519E	\$	22	46	1657
7.200E-04	\$	NOILASAP	\$	30	46	1658
8.160E-07	\$	NPP184AL	\$	32	46	1659
8.160E-07	\$	NPPAIRSL	\$	32	46	1660
1.000E-04	\$	NPS0001U	\$	32	46	1661
1.000E-04	\$	NPS0002U	\$	32	46	1662
7.200E-05	\$	NPS0107N	\$	13	46	1663
1.000E-03	\$	NPS0107U	\$	32	46	1664
7.200E-05	\$	NPS129SN	\$	13	46	1665
7.200E-05	\$	NPS130SN	\$	13	46	1666
7.200E-05	\$	NPS132SN	\$	13	46	1667
7.200E-05	\$	NPS166SM	\$	13	46	1668
7.200E-05	\$	NPS167SM	\$	13	46	1669
7.200E-05	\$	NPS169SM	\$	13	46	1670
7.200E-05	\$	NPS170SN	\$	13	46	1671
7.200E-05	\$	NPS171SN	\$	13	46	1672
7.200E-05	\$	NPS173SN	\$	13	46	1673
1.000E-03	\$	NPSSAC1U	\$	32	46	1674
2.640E-05	\$	NPV0103E	\$	21	46	1675
2.640E-05	\$	NPV0142E	\$	21	46	1676
2.640E-05	\$	NPV5756E	\$	22	46	1677
2.640E-05	\$	NPVAREGE	\$	21	1	1678
1.000E-04	\$	NPVAREGU	\$	32	1	1679

2.640E-05	\$	NPVBREGE	\$	21	1	1680
1.000E-04	\$	NPVBREGU	\$	32	1	1681
1.248E-04	\$	NRCBAIRL	\$	30	46	1682
1.248E-04	\$	NRCSAIRL	\$	30	46	1683
4.800E-05	\$	NRV3181D	\$	21	7	1684
4.800E-05	\$	NRV3182D	\$	21	7	1685
4.800E-05	\$	NRV3289D	\$	21	7	1686
4.080E-05	\$	NSV1131E	\$	31	46	1687
4.080E-05	\$	NSV1132E	\$	31	46	1688
4.080E-05	\$	NSV1134E	\$	31	46	1689
4.080E-05	\$	NSV1531E	\$	31	46	1690
7.200E-05	\$	NTEIAC1N	\$	13	46	1691
7.200E-05	\$	NTEIAC2N	\$	13	46	1692
7.200E-05	\$	NTESAC1N	\$	13	46	1693
2.040E-05	\$	NTKBOTAL	\$	23	1	1694
2.040E-05	\$	NTKBOTBL	\$	23	1	1695
8.160E-06	\$	NTV0310E	\$	21	46	1696
8.160E-06	\$	NTV0311C	\$	21	46	1697
8.160E-06	\$	NTV1021E	\$	21	46	1698
8.160E-06	\$	NTV1022E	\$	21	46	1699
8.160E-06	\$	NTV1035C	\$	21	46	1700
8.160E-06	\$	NTV1036C	\$	21	46	1701
8.160E-06	\$	NTV1038C	\$	21	46	1702
8.160E-06	\$	NTV1039C	\$	21	46	1703
8.160E-06	\$	NTV1042C	\$	21	46	1704
8.160E-06	\$	NTV1043C	\$	21	46	1705
8.160E-06	\$	NTV1045C	\$	21	46	1706
8.160E-06	\$	NTV1051C	\$	21	46	1707
8.160E-06	\$	NTV1052C	\$	21	46	1708
8.160E-06	\$	NTV1053C	\$	21	46	1709
8.160E-06	\$	NTV1054E	\$	21	46	1710
1.200E-05	\$	NTV1057D	\$	21	46	1711
8.160E-06	\$	NTV1093E	\$	22	46	1712
8.160E-06	\$	NTV1094E	\$	22	46	1713
8.160E-06	\$	NTV1191C	\$	21	46	1714
8.160E-06	\$	NTV1192C	\$	21	46	1715
8.160E-06	\$	NTV1195C	\$	21	46	1716
1.200E-05	\$	NTV1201D	\$	22	46	1717
1.200E-05	\$	NTV1251D	\$	22	46	1718
1.200E-05	\$	NTV1252D	\$	22	46	1719
1.200E-05	\$	NTV1252D	\$	22	46	1720
1.200E-05	\$	NTV1253D	\$	22	46	1721
1.200E-05	\$	NTV1254D	\$	22	46	1722
8.160E-06	\$	NTV1261C	\$	22	46	1723
8.160E-06	\$	NTV1262C	\$	22	46	1724
8.160E-06	\$	NTV1861C	\$	21	46	1725
8.160E-06	\$	NTV1862C	\$	21	46	1726
8.160E-06	\$	NTV5192E	\$	22	46	1727
8.160E-06	\$	NTV5193E	\$	22	46	1728
1.200E-05	\$	NTVABDVD	\$	21	1	1729
1.200E-05	\$	NTVAVNTD	\$	21	1	1730
1.200E-05	\$	NTVBBDVD	\$	21	1	1731
8.160E-06	\$	NTVBVA1E	\$	21	1	1732
8.160E-06	\$	NTVBVA2E	\$	21	1	1733
8.160E-06	\$	NTVBVB1E	\$	21	1	1734
8.160E-06	\$	NTVBVB2E	\$	21	1	1735
1.200E-05	\$	NTVBVNTD	\$	21	1	1736
8.160E-06	\$	NTVC1OTE	\$	21	7	1737
8.160E-06	\$	NTVC2OTE	\$	21	7	1738
8.160E-06	\$	NTVNV11E	\$	21	1	1739

8.160E-06	\$	NTVNV12E	\$	21	1	1740
1.000E+00	\$	SLOCA	\$	32	1	1741
4.000E-03	\$	T11ER	\$	32	1	1742
1.000E-01	\$	T13ER	\$	32	1	1743
1.000E+00	\$	T2H33R1	\$	32	1	1744
1.000E+00	\$	T2H33R3	\$	32	1	1745
1.000E+00	\$	V3R	\$	32	1	1746
3.500E-03	\$	VCRCMP1F	\$	16	32	1747
5.000E-05	\$	VCRCMP1G	\$	32	32	1748
1.600E-07	\$	VCRCMP1W	\$	32	32	1749
3.500E-03	\$	VCRCMP2F	\$	16	32	1750
5.000E-05	\$	VCRCMP2G	\$	32	32	1751
1.600E-07	\$	VCRCMP2W	\$	32	32	1752
1.420E-06	\$	VCV5751C	\$	22	31	1753
1.420E-06	\$	VCV5752C	\$	22	31	1754
1.420E-06	\$	VCV5753C	\$	22	31	1755
1.420E-06	\$	VCV6161C	\$	22	31	1756
1.420E-06	\$	VCV6162C	\$	22	31	1757
6.000E-05	\$	VDYDRI1E	\$	30	31	1758
6.000E-05	\$	VDYDRI2E	\$	30	31	1759
6.800E-06	\$	VGVS5571E	\$	22	31	1760
6.800E-06	\$	VGVS5572E	\$	22	31	1761
6.800E-06	\$	VGVS5573E	\$	22	31	1762
6.800E-06	\$	VGVS5861E	\$	22	31	1763
6.800E-06	\$	VGVS5861E	\$	22	31	1764
1.000E-03	\$	VGVS5861V	\$	32	31	1765
1.000E-03	\$	VGVS5861V	\$	32	31	1766
6.800E-06	\$	VGVS5862E	\$	22	31	1767
6.800E-06	\$	VGVS5862E	\$	22	31	1768
1.000E-03	\$	VGVS5862V	\$	32	31	1769
1.000E-03	\$	VGVS5862V	\$	32	31	1770
6.800E-06	\$	VGVS6361E	\$	22	31	1771
6.800E-06	\$	VGVS6362E	\$	22	31	1772
2.200E-06	\$	VPV6341E	\$	19	31	1773
2.200E-06	\$	VPV6342E	\$	19	31	1774
1.040E-05	\$	VRCSAR1L	\$	25	31	1775
1.040E-05	\$	VRCSAR2L	\$	25	31	1776
1.040E-05	\$	VRCSAR3L	\$	25	31	1777
4.000E-06	\$	VRV6351D	\$	22	31	1778
4.000E-06	\$	VRV6352D	\$	22	31	1779
6.800E-07	\$	VTVO632E	\$	22	29	1780
6.800E-07	\$	VTV5582E	\$	22	31	1781
6.800E-07	\$	VTV5881E	\$	22	31	1782
6.800E-07	\$	VTV5891E	\$	22	31	1783
6.800E-07	\$	VTV5893E	\$	22	31	1784
6.800E-07	\$	VTV5901E	\$	22	31	1785
6.800E-07	\$	VTV5902E	\$	22	31	1786
6.800E-07	\$	VTV5981E	\$	22	31	1787
6.800E-07	\$	VTV5982E	\$	22	31	1788
6.800E-07	\$	VTV6331E	\$	22	31	1789
6.800E-07	\$	VTV6332E	\$	22	31	1790
3.000E-05	\$	XCD1203D	\$	3	8	1791
3.000E-05	\$	XCD1B33D	\$	3	8	1792
1.000E-04	\$	XMESF1P	\$	31	7	1793
1.000E-04	\$	XMESF2P	\$	31	7	1794
0.000E-01	\$	XMMSV31P	\$	31	7	1795
0.000E-01	\$	XMMSV32P	\$	31	7	1796
0.000E-01	\$	XMMSV33P	\$	31	7	1797
0.000E-01	\$	XMMSV34P	\$	31	7	1798
0.000E-01	\$	XMMSV35P	\$	31	7	1799

0.000E-01	\$	XMMSV36P	\$	31	7	1800
0.000E-01	\$	XMMSV37P	\$	31	7	1801
0.000E-01	\$	XMMSV38P	\$	31	7	1802
0.000E-01	\$	XMMSV41P	\$	31	7	1803
0.000E-01	\$	XMMSV42P	\$	31	7	1804
0.000E-01	\$	XMMSV43P	\$	31	7	1805
0.000E-01	\$	XMMSV44P	\$	31	7	1806
0.000E-01	\$	XMMSV45P	\$	31	7	1807
0.000E-01	\$	XMMSV46P	\$	31	7	1808
0.000E-01	\$	XMMSV47P	\$	31	7	1809
0.000E-01	\$	XMMSV48P	\$	31	7	1810
1.000E-03	\$	XOP3H33V	\$	32	1	1811
1.000E+00	\$	XOPHLD1Q	\$	42	1	1812
1.000E+00	\$	XOPHLD2Q	\$	42	1	1813
1.000E+00	\$	XOPHLD3Q	\$	42	1	1814
1.000E+00	\$	XOPLLD1Q	\$	42	1	1815
1.000E+00	\$	XOPLLD2Q	\$	42	1	1816
1.000E+00	\$	XOPLLD3Q	\$	42	1	1817
3.000E-05	\$	XPBLOC1A	\$	13	29	1818
3.000E-05	\$	XPBLOC2A	\$	13	29	1819
3.000E-05	\$	XPBLOC3A	\$	13	29	1820
3.000E-05	\$	XPBLOC4A	\$	13	40	1821
3.000E-05	\$	XPBLOC5A	\$	13	40	1822
3.000E-05	\$	XPBLOC6A	\$	13	40	1823
3.000E-05	\$	XPBREM1A	\$	13	5	1824
3.000E-05	\$	XPBREM2A	\$	13	5	1825
3.000E-05	\$	XPBREM3A	\$	13	5	1826
3.000E-05	\$	XPBREM4A	\$	13	5	1827
3.000E-05	\$	XPBREM5A	\$	13	5	1828
3.000E-05	\$	XPBREM6A	\$	13	5	1829
3.000E-04	\$	XRL13K3H	\$	2	8	1830
3.000E-04	\$	XRL1K33H	\$	2	8	1831
3.000E-04	\$	XRL2K33H	\$	2	8	1832
3.000E-04	\$	XRL793AH	\$	2	53	1833
3.000E-04	\$	XRL793BH	\$	2	53	1834
3.000E-04	\$	XRL793CH	\$	2	53	1835
3.000E-04	\$	XRL793DH	\$	2	53	1836
3.000E-04	\$	XRL793EH	\$	2	53	1837
3.000E-04	\$	XRL793FH	\$	2	53	1838
3.000E-04	\$	XRL803AH	\$	2	53	1839
3.000E-04	\$	XRL803BH	\$	2	53	1840
3.000E-04	\$	XRL803CH	\$	2	53	1841
3.000E-04	\$	XRL803DH	\$	2	53	1842
3.000E-04	\$	XRL803EH	\$	2	53	1843
3.000E-04	\$	XRL803FH	\$	2	53	1844
3.000E-04	\$	XRL88K3H	\$	2	8	1845
3.000E-04	\$	XRL89K3H	\$	2	8	1846
3.000E-04	\$	XRL9033H	\$	2	8	1847
0.000E-01	\$	XRLRA1AJ	\$	2	8	1848
0.000E-01	\$	XRLRA1BJ	\$	2	8	1849
0.000E-01	\$	XRLRA1CJ	\$	2	8	1850
0.000E-01	\$	XRLRA1DJ	\$	2	8	1851
0.000E-01	\$	XRLRA1EJ	\$	2	8	1852
0.000E-01	\$	XRLRA1FJ	\$	2	8	1853
3.900E-07	\$	XRTS	\$	12	7	1854
6.200E-04	\$	XSV5081A	\$	31	7	1855
6.200E-04	\$	XSV5082A	\$	31	7	1856
6.200E-04	\$	XSV5083A	\$	31	7	1857
6.200E-04	\$	XSV5611A	\$	31	7	1858
6.200E-04	\$	XSV5612A	\$	31	7	1859

6.200E-04	\$	XSV5613A	\$		31	7		1860
0.0	\$	ASLEVEL1	\$	1.0	0	0		1861
0.0	\$	ASLEVEL2	\$		0	0		1862
1.000	\$	ASLEVEL3	\$		0	0		1863
1.000	\$	AHOUSE1	\$		0	0		1864
1.000	\$	GHOUSE3	\$		0	0		1865
1.000	\$	GHOUSE4	\$		0	0		1866
1.000	\$	GHOUSE5	\$		0	0		1867
0.000	\$	CHOUSE9	\$		0	0		1868
0.0	\$	KHOUSE1	\$		0	0		1869
1.000	\$	THOUSE	\$		0	0		1870
1.000	\$	NOTLLOCA	\$		0	0		1871
1.000	\$	NOSLOCA	\$		0	0		1872
2.730E-03	\$	PT-VAL-A2	\$		0	0		1873
0.0	\$	BLDG-105	\$		34	1		1874
0.0	\$	BLDG-109	\$		35	1		1875
0.0	\$	PRIM-SHIELD	\$		36	1		1876
0.0	\$	BLDG-181	\$		37	1		1877
0.0	\$	SILO	\$		38	1		1878
0.0	\$	DEMIN-TNK	\$		39	1		1879
0.0	\$	BLDG-182 **TEMP**	\$		37	1		1880
0.0	\$	SLOCA-PIPES	\$		43	1		1881
0.0	\$	LLOCA-PIPES	\$		45	1		1882
0.0	\$	BLDG 182	\$		37	1		1883
0.0	\$	CCF 13.8KV BUS	\$		46	32	1	1884
0.0	\$	CCF 4KV BUS	\$		47	32	1	1885

DESCRIPTIONS OF BASIC EVENTS

B.E. names

Descriptions:

A002MOD.....	INSUFF. FLOW OF WATER OUT OF THE EDT
A004DEL.....	A004MOD REMOVED FROM ECCS TREE: INCLUDED IN SEQUENCE LOGIC
A009MOD.....	V4-1 FAILURES
A010MOD.....	V4-2 FAILURES
A011MOD.....	V4-3 FAILURES
A012MOD.....	V4-4 FAILURES
A013MOD.....	V4-5 FAILURES
A014MOD.....	V4-6 FAILURES
A015MOD.....	V4-7 FAILURES
A016MOD.....	V4-8 FAILURES
A018MOD.....	V4-1 AIR CYLINDER FAILS
A019MOD.....	VARIOUS V4-1 RELATED HPAS VALVE FAILURES
A021MOD.....	V4-2 AIR CYLINDER FAILS
A022MOD.....	VARIOUS V4-2 RELATED HPAS VALVE FAILURES
A024MOD.....	V4-3 AIR CYLINDER FAILS
A025MOD.....	VARIOUS V4-3 RELATED HPAS VALVE FAILURES
A027MOD.....	V4-4 AIR CYLINDER FAILS
A028MOD.....	VARIOUS V4-4 RELATED HPAS VALVE FAILURES
A030MOD.....	V4-5 AIR CYLINDER FAILS
A031MOD.....	VARIOUS V4-5 RELATED HPAS VALVE FAILURES
A033MOD.....	V4-6 AIR CYLINDER FAILS
A034MOD.....	VARIOUS V4-6 RELATED HPAS VALVE FAILURES
A036MOD.....	V4-7 AIR CYLINDER FAILS
A037MOD.....	VARIOUS V4-7 RELATED HPAS VALVE FAILURES
A039MOD.....	V4-8 AIR CYLINDER FAILS
A040MOD.....	VARIOUS V4-8 RELATED HPAS VALVE FAILURES
A069MOD.....	V3-1 FAILURES
A070MOD.....	V3-2 FAILURES
A071MOD.....	V3-3 FAILURES
A072MOD.....	V3-4 FAILURES
A075MOD.....	CV'S 60-1 & 60-2
A082MOD.....	V3-5 FAILURES
A083MOD.....	V3-6 FAILURES
A084MOD.....	V3-7 FAILURES
A085MOD.....	V3-8 FAILURES
A087MOD.....	SUCTION PATH TO HLDPS FAILS (VALVE/PIPE
A122MOD.....	SILO FAILS OR RETURN VALVE FAILS
A210MOD.....	V3-2 AIR CYLINDER FAILS
A211MOD.....	VARIOUS V3-2 RELATED HPAS VALVE FAILURES
A218MOD.....	V3-1 AIR CYLINDER FAILS
A219MOD.....	VARIOUS V3-1 RELATED HPAS VALVE FAILURES
A240MOD.....	V3-3 AIR CYLINDER FAILS
A241MOD.....	VARIOUS V3-3 RELATED HPAS VALVE FAILURES
A270MOD.....	V3-4 AIR CYLINDER FAILS
A271MOD.....	VARIOUS V3-4 RELATED HPAS VALVE FAILURES
A300MOD.....	V3-5 AIR CYLINDER FAILS
A301MOD.....	VARIOUS V3-5 RELATED HPAS VALVE FAILURES
A330MOD.....	V3-6 AIR CYLINDER FAILS
A331MOD.....	VARIOUS V3-6 RELATED HPAS VALVE FAILURES
A360MOD.....	V3-7 AIR CYLINDER FAILS
A361MOD.....	VARIOUS V3-7 RELATED HPAS VALVE FAILURES
A390MOD.....	V3-8 AIR CYLINDER FAILS
A391MOD.....	VARIOUS V3-8 RELATED HPAS VALVE FAILURES
A508AMOD.....	HLDPS #1 SELF COOLING VALVE FAILURES

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
A508BMOD.....	COMMON VALVES TO HLDP #1 COOLING FAIL
A508DMOD.....	HLDP #2 SELF COOLING VALVE FAILURES
A508EMOD.....	COMMON VALVES TO HLDP #2 COOLING FAIL
A508FMOD.....	HLDP #3 SELF COOLING VALVE FAILURES
A508GMOD.....	COMMON VALVES TO HLDP #3 COOLING FAIL
A530MOD.....	VALVES ON LLDP #3 FAIL
A613MOD.....	HLDP #1 AIR SUPPLY VALVES FAIL
A614MOD.....	HLDP #2 START AIR SUPPLY VALVES FAIL
A615MOD.....	HLDP #3 AIR START SUPPLY VALVES FAIL
A616MOD.....	LLDP #1 START AIR VALVES FAIL
A617MOD.....	LLDP #2 START AIR SUPPLY VALVES FAIL
A621MOD.....	VALVES FROM AIR RECEIVER #1 FAIL
AAG19--D.....	V-19 FAILS TO REMAIN CLOSED
AAG3-1-C.....	V3-1 FAILS TO REMAIN OPEN
AAG3-2-C.....	V3-2 FAILS TO REMAIN OPEN
AAG3-3-C.....	V3-3 FAILS TO REMAIN OPEN
AAG3-4-C.....	V3-4 FAILS TO REMAIN OPEN
AAG3-5-C.....	V3-5 FAILS TO REMAIN OPEN
AAG3-6-C.....	V3-6 FAILS TO REMAIN OPEN
AAG3-7-C.....	V3-7 FAILS TO REMAIN OPEN
AAG3-8-C.....	V3-8 FAILS TO REMAIN OPEN
AAG4-1-A.....	V4-1 FAILS TO OPEN ON DEMAND
AAG4-2-A.....	V4-2 FAILS TO OPEN ON DEMAND
AAG4-3-A.....	V4-3 FAILS TO OPEN ON DEMAND
AAG4-4-A.....	V4-4 FAILS TO OPEN ON DEMAND
AAG4-5-A.....	V4-5 FAILS TO OPEN ON DEMAND
AAG4-6-A.....	V4-6 FAILS TO OPEN ON DEMAND
AAG4-7-A.....	V4-7 FAILS TO OPEN ON DEMAND
AAG4-8-A.....	V4-8 FAILS TO OPEN ON DEMAND
AAGV3CMA.....	COMMON CAUSE FAILURE OF A SIDE OF V3S TO OPEN ON DEMAND
AAGV4CMA.....	COMMON CAUSE FAILURE OF A SIDE OF V4S TO OPEN ON DEMAND
AARHL1-L.....	HLDP 1 ACCUMULATOR FAILS TO DELIVER AIR
AARHL2-L.....	HLDP 2 ACCUMULATOR FAILS TO DELIVER AIR
AARHL3-L.....	HLDP 3 ACCUMULATOR FAILS TO DELIVER AIR
AARLL1-L.....	LLDP 1 ACCUMULATOR FAILS TO DELIVER AIR
AARLL2-L.....	LLDP 2 ACCUMULATOR FAILS TO DELIVER AIR
AARLL3-L.....	LLDP 3 ACCUMULATOR FAILS TO DELIVER AIR
ACV5001A.....	CAEV-500-1 FAILS TO OPEN ON DEMAND
ACV5002A.....	CAEV-500-2 FAILS TO OPEN ON DEMAND
ACV5035A.....	DOSV-503-5 FAILS TO OPEN ON DEMAND
ACV5036A.....	DOSV-503-6 FAILS TO OPEN ON DEMAND
ACV504-A.....	CAEV-504 FAILS OPEN ON DEMAND
ACV5041A.....	RWSV-504-1R FAILS TO OPEN ON DEMAND
ACV5042A.....	RWSV-504-2R FAILS TO OPEN ON DEMAND
ACV510-A.....	CAEV-510 FAILS TO OPEN ON DEMAND
ACV517-A.....	CAEV-517 FAILS TO OPEN ON DEMAND
ACV518-A.....	CAEV-518 FAILS TO OPEN ON DEMAND
ACV5543A.....	DOSV-554-3 FAILS TO OPEN ON DEMAND
ACV5544A.....	DOSV-554-4 FAILS TO OPEN ON DEMAND
ACV5545A.....	DOSV-554-5 FAILS TO OPEN ON DEMAND
ACV5621A.....	RWSV-562-1 FAILS TO OPEN ON DEMAND
ACV5622A.....	RWSV-562-2 FAILS TO OPEN ON DEMAND
ACV5623A.....	RWSV-562-3 FAILS TO OPEN ON DEMAND
ACV5731A.....	CAEV-573-1 FAILS TO OPEN ON DEMAND
ACV5732A.....	CAEV-573-2 FAILS TO OPEN ON DEMAND
ACV5733A.....	CAEV-573-3 FAILS OPEN ON DEMAND

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
ACV812-A.....	FWV-812 FAILS TO OPEN ON DEMAND
ACV820-A.....	DWV-820 FAILS TO OPEN ON DEMAND
ADPHL1-F.....	PUMP FAILS TO START #1
ADPHL1-G.....	HLDP 1 FAILS TO RUN
ADPHL2-F.....	PUMP FAILS TO START #2
ADPHL2-G.....	HLDP 2 FAILS TO RUN
ADPHL3-F.....	PUMP FAILS TO START #3
ADPHL3-G.....	HLDP 3 FAILS TO RUN
ADPLL1-F.....	LLDP #1 FAILS TO START
ADPLL1-G.....	LLDP 1 FAILS TO RUN
ADPLL2-F.....	LLDP #2 FAILS TO START
ADPLL2-G.....	LLDP 2 FAILS TO RUN
ADPLL3-F.....	LLDP #3 FAILS TO START
ADPLL3-G.....	LLDP 3 FAILS TO RUN
ADV5091C.....	LLDP #1 COOLING FAILS DUE TO RWV-509-1
ADV5092C.....	LLDP #2 COOLING FAILS DUE TO RWV-509-2
ADV515-C.....	LLDP #3 COOLING FAILS DUE TO RWV-515
AFV11--C.....	ANY TWO V-11'S FTRO (POST TRIP)
AFV5525C.....	RWSV-552-5 FAILS TO REMAIN OPEN
AFV5526C.....	RWSV-552-6 FAILS TO REMAIN OPEN
AFV5527C.....	RWSV-552-7 FAILS TO REMAIN OPEN
AGVF506C.....	FWV-506 FAILS TO REMAIN OPEN
AGVF507C.....	FWV-507 FAILS TO REMAIN OPEN
AGVF510C.....	FWV-510 FAILS TO REMAIN OPEN
AGVF511C.....	FWV-501-1 FAILS TO REMAIN OPEN
AGVF512C.....	FWV-501-2 FAILS TO REMAIN OPEN
AGVF611C.....	FWV-601-1 FAILS TO REMAIN OPEN
AGVF612C.....	FWV-601-2 FAILS TO REMAIN OPEN
AGVF613C.....	FWV-601-3 FAILS TO REMAIN OPEN
AHOUSE1.....	NEED FOR MANUAL START OF ECCS FOLLOWING SCRAM
AMT8161C.....	DWV-816-1 FAILS TO REMAIN OPEN
AMT8171A.....	FWV-817-1 FAILS TO OPEN
AMT8171C.....	FWV-817-1 FAILS TO REMAIN OPEN
AMT8171Q.....	OPERAT. FAILS TO OPEN FWV-817-1 IN 8 MIN.
AMV181-D.....	V18-1 FAIL TO REMAIN CLOSED
AMV182-D.....	V18-2 FAIL TO REMAIN CLOSED
AMV5541D.....	RWSV-554-1R FAILS TO REMAIN CLOSED
AOP2H33V.....	OPER. FAILS RESET BYPASS SW. 2H33 AFTER SCRAM & MANUAL ECCS
AOPHLDPQ.....	OPERATOR FAILS TO THROTTLE HLDP, 5 MIN.
APPECCSL.....	30 INCH HEADER RUPTURES
APV6141C.....	RWSV-614-1 FAILS TO REMAIN OPEN
ASLEVEL1.....	FIRST 23.3 MINUTES - 3 SOURCES OF WATER
ASLEVEL2.....	20 TO 85.5 MINUTES - 2 SOURCES OF WATER
ASLEVEL3.....	AFTER 85.5 MINUTES - 1 SOURCE OF WATER
ASV5021A.....	FWV-502-1 FAILS TO OPEN ON DEMAND
ASV5022A.....	FWV-502-2 FAILS TO OPEN ON DEMAND
ASV6001A.....	FWV-600-1 FAILS TO OPEN ON DEMAND
ASV6002A.....	FWV-600-2 FAILS TO OPEN ON DEMAND
ASV6003A.....	FWV-600-3 FAILS TO OPEN ON DEMAND
ATKH1EXL.....	HLDP #1 JACKET H2O EXPANSION TANK LEAKS
ATKH2EXL.....	HLDP #2 JACKET H2O EXPANSION TANK LEAKS
ATKH3EXL.....	HLDP #3 JACKET H2O EXPANSION TANK LEAKS
ATKL1EXL.....	LLDP #1 JACKET WATER EXPANSION TANK LEAK
ATKL2EXL.....	LLDP #2 JACKET WATER EXPANSION TANK LEAK
ATKL3EXL.....	LLDP #3 JACKET WATER EXPANSION TANK LEAK
ATKN01-L.....	DAY TANK #1 FAILS TO DELIVER FUEL

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
ATKN02-L.....	DAY TANK #2 FAILS TO DELIVER FUEL
ATKN03-L.....	DAY TANK #3 FAILS TO DELIVER FUEL
ATT5021A.....	JWV-502-1 FAILS TO OPEN
ATT5022A.....	JWV-502-2 FAILS TO OPEN
ATT510-A.....	JWV-510 FAILS TO OPEN
ATT5521A.....	JWV-552-1 FAILS TO OPEN
ATT5522A.....	JWV-552-2 FAILS TO OPEN
ATT5523A.....	JWV-552-3 FAILS TO OPEN
ATV502-C.....	DOSV-502 FAILS TO REMAIN OPEN
ATV516-C.....	CAEV-516 FAILS TO REMAIN OPEN
ATV5534C.....	CAEV-553-4 FAILS TO REMAIN OPEN
ATV5535C.....	CAEV-553-5 FAILS TO REMAIN OPEN
ATV6221C.....	RWSV-622-1 FAILS TO REMAIN OPEN
ATV8161C.....	DOSV-816-1 FAILS TO REMAIN OPEN
ATV8162C.....	DOSV-816-2 FAILS TO REMAIN OPEN
ATV8163C.....	DOSV-816-3 FAILS TO REMAIN OPEN
ATV8171C.....	DOSV-817-1 FAILS TO REMAIN OPEN
ATV8181C.....	DOSV-818-1 FAILS TO REMAIN OPEN
ATV8182C.....	DOSV-818-2 FAILS TO REMAIN OPEN
AXV516-C.....	LLDP #3 COOLING FAILS DUE TO RWV-516
AXV6201C.....	RWSV-620-1 FAILS TO REMAIN OPEN
C110MOD.....	GSCS PUMP 1 FAILS
C115MOD.....	GSCS PUMP 1 SEAL WATER DP INDIC FAILS
C117MOD.....	GSCS PUMP 1 LOW PRESS INDIC FAILS
C120MOD.....	GSCS PUMP 2 FAILS
C125MOD.....	GSCS PUMP 2 SEAL WATER DP INDIC FAILS
C127MOD.....	GSCS PUMP 2 LOW PRESS INDIC FAILS
C130MOD.....	GSCS PUMP 3 FAILS
C160MOD.....	GSCS PUMP 3 FAILS TO START
C161MOD.....	GSCS PUMP 3 UNAVAILABLE
C202MOD.....	GSCS MAKEUP LINES 1,2 CLOSED
C251MOD.....	GSCS SPILL CONTROL FAILS
C280MOD.....	GSCS MAKEUP CONTROL FAILS
C304MOD.....	GSCS HEX FLOW TEMP INSTRUMENTATION FAILS
C306MOD.....	GSCS HEX CRW TEMP INSTRUMENTATION FAILS
C310MOD.....	GSCS HEX 1 FAILS TO REMOVE HEAT
C320MOD.....	GSCS HEX 2 FAILS TO REMOVE HEAT
C330MOD.....	GSCS HEX 3 FAILS TO REMOVE HEAT
C340MOD.....	GSCS HEX 4 FAILS TO REMOVE HEAT
C410MOD.....	GSCS BULK TEMPERATURE CONTROL FAILS
C424MOD.....	GSCS TUBE FLOW INDICATION FAILS
C442MOD.....	GSCS PRESSURE LESS THAN 90 PSIG
C443MOD.....	N2 SUPPLY FAILS
C446MOD.....	GSCS PRESSURE TRANSMITTERS FAIL
C447MOD.....	GSCS PRESSURE CONTROL VALVES FAIL
C448MOD.....	GSCS RELIEF VALVES FAIL TO OPEN
C450MOD.....	NO INDICATION OF PRESS CONTROL FAILURE
C602MOD.....	ONCE-THRU FLOW TRIP INSTRUMENTATION FAILS
C604MOD.....	ONCE-THRU TEMP TRIP INSTRUMENTATION FAILS
C610MOD.....	ONCE-THRU MANUAL TRIP ANNUNCIATORS FAIL
CAG3011C.....	GSCS RET ISOL GCRV-301-1 FAILS TRO
CAG3081C.....	GSCS SUP ISOL GCSV-308-1 FAILS TRO
CAN00050.....	GSCS PUMP 1 BREAKER TRIP ANNUN #5 FAILS
CAN00060.....	GSCS PUMP 2 BREAKER TRIP ANNUN #6 FAILS
CAN04280.....	ONCE-THRU BYPASS ANNUN #428 FAILS
CAV2341C.....	RWS SUP RWSV-234-1 FAILS TO REMAIN OPEN

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
CAV2342C.....	RWS SUP RWSV-234-2 FAILS TO REMAIN OPEN
CCV0007C.....	GCDV-7 FAILS TO OPEN COMPLETELY WHEN REQUIRED
CCV2050C.....	RWS SUP RWSV-205 FAILS TO REMAIN OPEN
CCV2060C.....	SEAL WATER DWV-206 FAILS TO REMAIN OPEN
CCV3071B.....	GSCS PUMP 1 GCSV-303-1 FAILS TO CLOSE
CCV3072B.....	GSCS PUMP 2 GCSV-303-2 FAILS TO CLOSE
CDB0001C.....	GSCS CONTROL GCSV-1 FAILS TO REM OPEN
CDB3161A.....	RWS GCSV-316-1R FAILS TO OPEN
CDB3162A.....	RWS GCSV-316-2 FAILS TO OPEN
CDT0011A.....	GSCS DUMP GCDV-1-1R FAILS TO OPEN
CDT0011D.....	GSCS DUMP GCDV-1-1R FAILS TO REM CLOSED
CDT0012A.....	GSCS DUMP GCDV-1-2 FAILS TO OPEN
CDT0012D.....	GSCS DUMP GCDV-1-2 FAILS TO REM CLOSED
CDV3181B.....	GSCS SPILL L1 GCSV-318-1 FAILS TO CLOSE
CDV3182B.....	GSCS SPILL L2 GCSV-318-2 FAILS TO CLOSE
CFFBULK0.....	GSCS BULK FLOW ELEMENT FAILS
CFNORIFE.....	GSCS DUMP FLOW ORIFICE PLUGS
CFNORIFL.....	GSCS DUMP ORIFICE FAILS
CFR1400M.....	GSCS BULK FLOW REC RR-1400 FAILS HIGH
CFX0FT7M.....	GSCS BULK FLOW TRAN FT-7 FAILS HIGH
CHOUSE1.....	GSCS CONTROL, FAILURE INDICATION AVAILABLE
CHOUSE9.....	PROCESS TUBE FAILURE
CMT2041C.....	SEAL WATER DWV-204 FAILS TO REMAIN OPEN
CMT302RC.....	GSCS RET ISOL GCRV-302-1 FAILS TRO
CMT3071D.....	RECIRC GCSV-307-1 FAILS TO REMAIN CLOSED
CMT3151B.....	GSCS SPILL L1 GCSV-315-1 FAILS TO CLOSE
CMT3152B.....	GSCS SPILL L2 GCSV-315-2 FAILS TO CLOSE
COPBYPAR.....	OPER FAILS TO RESPOND TO BYPASS ANNUN
COPBYPQS.....	OPER BYPASSES ONCE-THRU FLOW, TEMP TRIP
COPFLNGV.....	GSCS TO PRIM SPECT FLNGE NOT INSTALLED
COPFLTRU.....	MISCAL: FLOW TRIP INSTRUMENTATION
COPMP3Q.....	OPER FAILS TO START GSCS PUMP 3
COPSBUSQ.....	OPER FAILS TO SWITCH TO POWERED BUS
COPSWITQ.....	GSCS ST LEVEL SELECTOR SWITCH MISALIGNED
COPTETRU.....	MISCAL: TEMP TRIP INSTRUMENTATION
COPTRIPQ.....	OPER FAILS TO TRIP TO ONCE-THRU COOLING
CPP105NL.....	GSCS PIPING FAILURE IN 105N
CPP109NL.....	GSCS PIPING FAILURE IN 109N
CPPBILCL.....	BOTTOM INSULATION LAYER DISCHARGE PIPING RUPTURES
CPPDUMPL.....	GSCS PIPING FAILURE IN ONCE-THRU HEADERS
CPRESS.....	UNSPECIFIED PRESSURE RISE IN GRAPHITE STACK
CRV0002D.....	GSCS DUMP REL GCDV-2 FAILS TO REM CLOSED
CSW0041D.....	GSCS PUMP 3 BUS SELECT SWIT SS-41 FAILS
CTK0000L.....	GSCS SURGE TANK RUPTURE
CTV2300C.....	RWS SUP RWSV-230 FAILS TO REMAIN OPEN
CTV3190D.....	GSCS TO PRIM GCRV-319 FAILS TO REM CLOSE
CTV3191B.....	GSCS SPILL L1 GCSV-319-1 FAILS TO CLOSE
CTV3192B.....	GSCS SPILL L2 GCSV-319-2 FAILS TO CLOSE
CZ2GCSDP.....	SUCCESSFUL HOT GSCS DUMP
DDA15.....	DA: AHR PUMP START SIGNAL FAILS
DDA25.....	DA: HPI PUMP START SIGNAL FAILS
E060/MOD.....	BFW PUMP #1 PATH FAILS EXCEPT H001,E090
E070/MOD.....	BFW PUMP #2 PATH FAILS EXCEPT H100,E090
E080/MOD.....	BFW PUMP #3 PATH FAILS EXCEPT E083,E090
E420/MOD.....	INSUFFICIENT FLOW FROM SLMU SYSTEM (ASSUMED)
E420MOD2.....	INSUFFICIENT FLOW FROM SLMU SYSTEM EXCEPT M300 AND M100

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
E430/MOD.....	INSUFFICIENT FLOW FROM AHR SYSTEM EXCEPT M200 AND N100
E705/MOD.....	DRIVE TURBINE #1 FAILS TO RUN EXCEPT E710B,G167,K100,E790
E706/MOD.....	DRIVE TURBINE #2 FAILS TO RUN EXCEPT E711B,G168,K100,E790
E707/MOD.....	DRIVE TURBINE #3 FAILS TO RUN EXCEPT E712B,G169,K100,E790
E708/MOD.....	DRIVE TURBINE #4 FAILS TO RUN EXCEPT E713B,G170,K100,E790
E709/MOD.....	DRIVE TURBINE #5 FAILS TO RUN EXCEPT E714B,G171,K100,E790
E710C/MO.....	LOSS OF DT #1 CONDENSATE TRAIN A EXCEPT H200,E710K
E710D/MO.....	LOSS OF DT #1 CONDENSATE TRAIN A EXCEPT H300,E710H
E711C/MO.....	LOSS OF DT #2 CONDENSATE TRAIN A EXCEPT H200,E711K
E711D/MO.....	LOSS OF DT #2 CONDENSATE TRAIN A EXCEPT H300,E711H
E712C/MO.....	LOSS OF DT #3 CONDENSATE TRAIN A EXCEPT H200,E712K
E712D/MO.....	LOSS OF DT #3 CONDENSATE TRAIN A EXCEPT H300,E712H
E713C/MO.....	LOSS OF DT #4 CONDENSATE TRAIN A EXCEPT H200,E713K
E713D/MO.....	LOSS OF DT #4 CONDENSATE TRAIN A EXCEPT H300,E713H
E714C/MO.....	LOSS OF DT #5 CONDENSATE TRAIN A EXCEPT H200,E714K
E714D/MO.....	LOSS OF DT #5 CONDENSATE TRAIN A EXCEPT H300,E714H
E836MOD.....	STEAM FLOW FROM SG 1A BLOCKED
E837MOD.....	STEAM FLOW FROM SG 1B BLOCKED
E838MOD.....	STEAM FLOW FROM SG 2A BLOCKED
E839MOD.....	STEAM FLOW FROM SG 2B BLOCKED
E840MOD.....	STEAM FLOW FROM SG 3A BLOCKED
E841MOD.....	STEAM FLOW FROM SG 3B BLOCKED
E842MOD.....	STEAM FLOW FROM SG 4A BLOCKED
E843MOD.....	STEAM FLOW FROM SG 4B BLOCKED
E844MOD.....	STEAM FLOW FROM SG 5A BLOCKED
E845MOD.....	STEAM FLOW FROM SG 5B BLOCKED
E990MOD.....	FAILURE TO PRESERVE THERMAL FLYWHEEL AFTER SCRAM
ECMDTHPP.....	DTH PRESSURE CONTROL INSTRUMENTATION FAILS
ECMDTLCP.....	DEARATOR TANK LEVEL CONTROLLER FAILS
ECMSTLCP.....	FAILURE OF SURGE TANK LEVEL CONTROL INSTRUMENTATION
ECV2420A.....	CONDV-242-10 FAILS TO OPEN ON DEMAND
ECV2422A.....	CONDV-242-2 FAILS TO OPEN ON DEMAND
ECV2424A.....	CONDV-242-4 FAILS TO OPEN ON DEMAND
ECV2426A.....	CONDV-242-6 FAILS TO OPEN ON DEMAND
ECV2428A.....	CONDV-242-8 FAILS TO OPEN ON DEMAND
EDT2431A.....	CONDV-243-1R FAILS TO OPEN
EDT2431C.....	CONDV-243-1R FAILS TO REMAIN OPEN
EDV2031A.....	MWV-203-1 FAILS TO OPEN
EDV2031C.....	MWV-203-1 FAILS TO REMAIN OPEN
EDV2032A.....	MWV-203-2 FAILS TO OPEN
EDV2032C.....	MWV-203-2 FAILS TO REMAIN OPEN
EDV2151D.....	CONDV-215-1 FAILS TO REMAIN CLOSED
EDV2152D.....	CONDV-215-2 FAILS TO REMAIN CLOSED
EDV6211C.....	HPV-6210-1 FAILS TO REMAIN OPEN
EDV6212C.....	HPV-6210-2 FAILS TO REMAIN OPEN
EDV6213C.....	HPV-6210-3 FAILS TO REMAIN OPEN
EDV6214C.....	HPV-6210-4 FAILS TO REMAIN OPEN
EGV2071C.....	HPV-207-1 24-INCH GATE VALVE FAILS TO REMAIN OPEN
EGV2072C.....	HPV-207-2 24-INCH GATE VALVE FAILS TO REMAIN OPEN
EGV2073C.....	HPV-207-3 24-INCH GATE VALVE FAILS TO REMAIN OPEN
EGV2074C.....	HPV-207-4 24 INCH GATE VALVE FAILS TO REMAIN OPEN
EGV2411C.....	CONDV-241-1 FAILS TO REMAIN OPEN
EGV2412C.....	CONDV-241-2 FAILS TO REMAIN OPEN
EGV2413C.....	CONDV-241-3 FAILS TO REMAIN OPEN
EGV2414C.....	CONDV-241-4 FAILS TO REMAIN OPEN
EGV2416C.....	CONDV-241-6 FAILS TO REMAIN OPEN

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
EGV2417C.....	CONDV-241-7 FAILS TO REMAIN OPEN
EGV2418C.....	CONDV-241-8 FAILS TO REMAIN OPEN
EGV2419C.....	CONDV-241-9 FAILS TO REMAIN OPEN
EGV4111C.....	CONDV-241-11 FAILS TO REMAIN OPEN
EGV4112C.....	CONDV-241-12 FAILS TO REMAIN OPEN
EGV4113C.....	CONDV-241-13 FAILS TO REMAIN OPEN
EGV4114C.....	CONDV-241-14 FAILS TO REMAIN OPEN
EGV4116C.....	CONDV-241-16 FAILS TO REMAIN OPEN
EGV4117C.....	CONDV-241-17 FAILS TO REMAIN OPEN
EGV4118C.....	CONDV-241-18 FAILS TO REMAIN OPEN
EGV4119C.....	CONDV-241-19 FAILS TO REMAIN OPEN
EGV4121C.....	CONDV-241-21 FAILS TO REMAIN OPEN
EGV4122C.....	CONDV-241-22 FAILS TO REMAIN OPEN
EGV4123C.....	CONDV-241-23 FAILS TO REMAIN OPEN
EGV4124C.....	CONDV-241-24 FAILS TO REMAIN OPEN
EPP1M-CL.....	PIPE BREAK BET 14-IN MMH HD & COND-1 HD
EPP2BHPL.....	28-IN HIGH PRESSURE LINE FAILS
EPP2M-CL.....	PIPE BREAK BET 14-IN MMH HD & COND-1 HD
EPPDEC2L.....	PIPE BR BET DIS-HD OF TRAIN ABC & COND-2
EPPFWPHL.....	PP BK BET BFWP INHD & DISHD OF TRAIN ABC
EPP1BH-L.....	30-IN INTER-BUILDING HEADER FAILS
EPPMSDTL.....	STEAM LINE BREAK DOWNSTREAM OF HPV-204,S TO TG OR DTS
EPPMSH-L.....	46-INCH MAIN STEAM HEADER FAILS
EPPPSDEL.....	20 IN PIPE BR BET BFWP SUCTION & DEAREATOR
ESCTURBP.....	LOSS OF TG/DT SURF COND FOR RX STEAM COOL ASSUMED, BOILER UP
ESTDT-1G.....	MECHANICAL FAILURE OF DRIVE TURBINE #1
ESTDT-2G.....	MECHANICAL FAILURE OF DRIVE TURBINE #2
ESTDT-3G.....	MECHANICAL FAILURE OF DRIVE TURBINE #3
ESTDT-4G.....	MECHANICAL FAILURE OF DRIVE TURBINE #4
ESTDT-5G.....	MECHANICAL FAILURE OF DRIVE TURBINE #5
ETKDESTL.....	DEAREATOR STOR TK RUPTURE
ETV1061C.....	GATE FPSV-106-1 FTRO, DEAR HT TO BFWP
ETV2052C.....	HPV-205-2 FAILS TO REMAIN OPEN
ETV2356C.....	HPV-235-6 FAILS TO REMAIN OPEN
ETV2441C.....	GATE CONDV-244-1 FTRO, BFWP TO COND 2
ETV2442C.....	GATE CONDV-244-2 FTRO, BFWP TO COND 2
ETV307-C.....	HPV-307 FAILS TO REMAIN OPEN
ETV308-C.....	HPV-308 FAILS TO REMAIN OPEN
F004MOD.....	DW TANK FAILS TO SUPPLY ADEQUATE WATER
F110MOD.....	HPAO FLUID COUPLING FAILS
F140MOD.....	HPAO DISCHARGE HEADER FAILS CLOSED
F210MOD.....	HPBO FLUID COUPLING FAILS
F240MOD.....	HPBO DISCHARGE HEADER FAILS CLOSED
F301MOD.....	HPAS FAILS TO START WHEN REQUIRED
F310MOD.....	HPAS FLUID COUPLING FAILS
F340MOD.....	HPAS DISCHARGE HEADER FAILS
F401MOD.....	HPBS FAILS TO START WHEN REQUIRED
F410MOD.....	HPBS FLUID COUPLING FAILS
F440MOD.....	HPBS DISCHARGE HEADER FAILS CLOSED
F503MOD.....	LOW FLOW INSTRUMENTATION FAILS
F802MOD.....	HPIP DP SIGNAL FAILS HIGH
F911MOD.....	B/U AIR REGULATORS FAIL TO REMAIN OPEN
FCBHPASK.....	HPAS A BUS BREAKER FAILS
FCBHPBSK.....	HPBS B BUS BREAKER FAILS
FCV579UC.....	HPI OPER PUMP CACV-579 FAILS TO REM OPEN
FCV579WC.....	HPI S/B PUMP CACV-579 FAILS TO REM OPEN

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
FCVB79UA.....	HPI OPER PUMP CACV-579 FAILS TO OPEN
FCVB79WA.....	HPI S/B PUMP CACV-579 FAILS TO OPEN
FFDHPAOP.....	HPAO FLUID COUPLING FAILS
FFDHPASP.....	HPAS FLUID COUPLING FAILS
FFDHPBOP.....	HPBO FLUID COUPLING FAIL
FFDHPBSP.....	HPBS FLUID COUPLING FAILS
FFV5511C.....	DW SUPPLY DWV-551-1 FAILS TO REMAIN OPEN
FFV5512C.....	DW SUPPLY DWV-551-2 FAILS TO REMAIN OPEN
FFV553UC.....	HPAO SUCTION DWV-553 FAILS TO REMAIN OP
FFV553VC.....	HPBO SUCTION DWV-553 FAILS TO REMAIN OP
FFV553WC.....	HPAS SUCTION DWV-553 FAILS TO REMAIN OP
FFV553XC.....	HPBS SUCTION DWV-553 FAILS TO REMAIN OP
FGV5583C.....	DIESEL B/U AIR CAEV-558-13 FAILS TRO
FGV5593C.....	DIESEL B/U AIR CAEV-559-3 FAILS TRO
FMPAOPUF.....	HPAO AUX OIL PUMP AUTOSTART FAILS
FMPAOPUG.....	HPAO AUX OIL PUMP FAILS TO RUN
FMPAOPVF.....	HPBO AUX OIL PUMP AUTOSTART FAILS
FMPAOPVG.....	HPBO AUX OIL PUMP FAILS TO RUN
FMPAOPWF.....	HPAS AUX OIL PUMP AUTOSTART FAILS
FMPAOPWG.....	HPAS AUX OIL PUMP FAILS TO RUN
FMPAOPXF.....	HPBS AUX OIL PUMP AUTOSTART FAILS
FMPAOPXG.....	HPBS AUX OIL PUMP FAILS TO RUN
FMPHPAOG.....	HPAO FAILS TO CONTINUE RUNNING
FMPHPASG.....	HPAS FAILS TO CONTINUE RUNNING
FMPHPBOG.....	HPBO FAILS TO CONTINUE RUNNING
FMPHPBSG.....	HPBS FAILS TO CONTINUE RUNNING
FMPOLPUG.....	HPAO OIL PUMP FAILS TO RUN
FMPOLPVG.....	HPBO OIL PUMP FAILS TO RUN
FMPOLPWG.....	HPAS OIL PUMP FAILS TO RUN
FMPOLPXG.....	HPBS OIL PUMP FAILS TO RUN
FOPHPDAV.....	HPI DISCONT ACTION SIGNAL BYPASSED
FOPHPIPU.....	MISCALIBRATION: HPIP DP INSTRUMENTATION
FOPHPBBR.....	OPERATOR FAILS TO START HPI S/B PUMPS
FPPHPIPL.....	HPIP MAIN HEADER FAILS
FTKOLRUL.....	HPAO OIL RESERVOIR FAILS
FTKOLRVL.....	HPBO OIL RESERVOIR FAILS
FTKOLRWL.....	HPAS OIL RESERVOIR FAILS
FTKOLRXL.....	HPBS OIL RESERVOIR FAILS
FTV0611C.....	DIESEL B/U AIR CASV-611 FAILS TO REM OP
FTV0683C.....	DIESEL B/U AIR SAV-683 FAILS TO REM OPEN
FTV0685C.....	DIESEL B/U AIR SAV-685 FAILS TO REM OPEN
FTV5513C.....	DIESEL B/U AIR CAEV-551-3 FAILS TRO
FTV577UC.....	HPI OPER PUMP CACV-577 FAILS TO REM OPEN
FTV577WC.....	HPI S/B PUMP CACV-577 FAILS TO REM OPEN
FTV599UC.....	HPI OPER PUMP CASV-599 FAILS TO REM OPEN
FTV600WC.....	HPI S/B PUMP CASV-600 FAILS TO REM OPEN
FTVC77UC.....	HPAO DIESEL B/U AIR CACV-577 FAILS TRO
FTVC77WC.....	HPAS DIESEL B/U AIR CACV-577 FAILS TRO
G003MOD.....	PRIMARY SYSTEM PRESSURE CONTROL EXCEPT N103
G035/MOD.....	CELL #1 FAILS ON PRIMARY SIDE EXCEPT G090,G102,G167
G037/MOD.....	CELL #2 FAILS ON PRIMARY SIDE EXCEPT G092,G103,G168
G039/MOD.....	CELL #3 FAILS ON PRIMARY SIDE EXCEPT G094,G104,G169
G041/MOD.....	CELL #4 FAILS ON PRIMARY SIDE EXCEPT G096,G105,G170
G043/MOD.....	CELL #5 FAILS ON PRIMARY SIDE EXCEPT G098,G106,G171
G121AMOD.....	NO START SIGNAL TO PONY MOTOR #1 EXCEPT J200
G121BMOD.....	NO START SIGNAL TO PONY MOTOR #2 EXCEPT J200

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
G121CMOD.....	NO START SIGNAL TO PONY MOTOR #3 EXCEPT J200
G121DMOD.....	NO START SIGNAL TO PONY MOTOR #4 EXCEPT J200
G121EMOD.....	NO START SIGNAL TO PONY MOTOR #5 EXCEPT J200
G155/MOD.....	INSUFF HEAT REMOVAL THRU SG 1A EXCEPT G300,E020,N103,E836MOD
G156/MOD.....	INSUFF HEAT REMOVAL THRU SG 1B EXCEPT G300,E020,N103,E837MOD
G157/MOD.....	INSUFF HEAT REMOVAL THRU SG 2A EXCEPT G300,E020,N103,E838MOD
G158/MOD.....	INSUFF HEAT REMOVAL THRU SG 2B EXCEPT G300,E020,N103,E839MOD
G159/MOD.....	INSUFF HEAT REMOVAL THRU SG 3A EXCEPT G300,E020,N103,E840MOD
G160/MOD.....	INSUFF HEAT REMOVAL THRU SG 3B EXCEPT G300,E020,N103,E841MOD
G161/MOD.....	INSUFF HEAT REMOVAL THRU SG 4A EXCEPT G300,E020,N103,E842MOD
G162/MOD.....	INSUFF HEAT REMOVAL THRU SG 4B EXCEPT G300,E020,N103,E843MOD
G163/MOD.....	INSUFF HEAT REMOVAL THRU SG 5A EXCEPT G300,E020,N103,E844MOD
G164/MOD.....	INSUFF HEAT REMOVAL THRU SG 5B EXCEPT G300,E020,N103,E845MOD
G240/MOD.....	LOSS OF A BUS POWER TO BAILEY INSTR PANELS EXCEPT H001
G241/MOD.....	LOSS OF B BUS POWER TO BAILEY INSTR PANELS EXCEPT H100
G301/MOD.....	LOSS POWER TO 120VAC TO L&M INSTR PANEL MAX-1-A EXCEPT H100
GAG2191D.....	CONTROL VALVE BDV-219-1 FTRC, 4 INCH PRIM/RY SPILL VALVE
GAG2192D.....	CONTROL VALVE BDV-219-2 FTRC, 4 IN PRIMAR SPILL VALVE
GAG2531A.....	VALVE IWV-253-1 FTO, 8 IN. PCS MAKE-UP VALVE
GAG2532A.....	VALVE IWV-253-2 FTO, 8 IN. PCS MAKE-UP VALVE
GBT8A1LK.....	240 VDC BATTERY WILL DISCH IN < 1 HR, FAILURE ASSUMED
GBUACCRI.....	LOSS OF 120 VAC CK PANEL BUS
GBUACEAI.....	LOSS OF 120 VAC EA PANEL BUS
GBLMAXBI.....	FAILURE OF INSTRUMENT PANEL MAX-1-B BUS
GC83-65H.....	CIRCUIT BREAKER CB3-65 FAILS OPEN
GCB8R/CH.....	FAILURE OF RECTIFIER/CHARGER CIRCUIT BREAKER
GCP1LB1G.....	MAIN LUBE OIL TRANSFER PUMP #1A FAILS
GCP1LB2G.....	MAIN LUBE OIL TRANSFER PUMP #1B FAILS
GCP2LB1G.....	MAIN LUBE OIL TRANSFER PUMP #2A FAILS
GCP2LB2G.....	MAIN LUBE OIL TRANSFER PUMP #2B FAILS
GCP3LB1G.....	MAIN LUBE OIL TRANSFER PUMP #3A FAILS
GCP3LB2G.....	MAIN LUBE OIL TRANSFER PUMP #3B FAILS
GCP4LB1G.....	MAIN LUBE OIL TRANSFER PUMP #4A FAILS
GCP4LB2G.....	MAIN LUBE OIL TRANSFER PUMP #4B FAILS
GCP5LB1G.....	MAIN LUBE OIL TRANSFER PUMP #5A FAILS
GCP5LB2G.....	MAIN LUBE OIL TRANSFER PUMP #5B FAILS
GEP202-M.....	E/P TRANSDUCER REL-202, FAILS HIGH
GEP2021M.....	P/E TRANSDUCER LT-202-1 FAILS HIGH
GEP203-M.....	E/P TRANSDUCER REL-203, FAILS HIGH
GEP2031M.....	P/E TRANSDUCER LT-203-1 FAILS HIGH
GEP244-M.....	E/P TRANSDUCER REL-244, FAILS HIGH
GEP248-M.....	E/P TRANSDUCER REL-248, FAILS HIGH
GEP2721M.....	P/E TRANSDUCER LT-272-1 FAILS HIGH
GFC234-M.....	FLOW CONTROLLER FC-234, FAILS HIGH
GFC236-M.....	FLOW CONTROLLER FC-236, FAILS HIGH
GFM001-F.....	PM FAN #1 FAILS TO START (WITHIN 15 MIN)
GFM002-F.....	PM FAN #2 FAILS TO START (WITHIN 15 MIN)
GFM003-F.....	PM FAN #3 FAILS TO START (WITHIN 15 MIN)
GFM004-F.....	PM FAN #4 FAILS TO START (WITHIN 15 MIN)
GFM005-F.....	PM FAN #5 FAILS TO START (WITHIN 15 MIN)
GFUACCRH.....	FAILURE OF INSTR PANEL CR FUSE
GFUACEAH.....	FAILURE OF INSTR PANEL EA FUSE
GFUMAXBH.....	FAILURE OF INSTR PANEL MAX-1-B FUSE
GHOUSE2.....	FIRST HALF HOUR - NUCLEAR STEAM TO TURBINES
GHOUSE3.....	AFTER FIRST HALF HOUR - BOILER STEAM TO TURBINES
GHOUSE4.....	STEAMING MODE, 0-3 HOUR OPERATING MODE FLAG

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
GHOUSE5.....	WATER TO WATER OPERATING MODE (3-24 HOUR) FLAG
GIVSTATH.....	FAILURE OF STATIC INVERTER
GLX202-M.....	LEVEL TRANSMITTER LT-202 FAILS HIGH
GLX203-M.....	LEVEL TRANSMITTER LT-203 FAILS HIGH
GLX272-M.....	LEVEL TRANSMITTER LT-272 FAILS HIGH
GPM001-F.....	PONY MOTOR #1 FAILS TO START
GPM001-G.....	PONY MOTOR #1 FAILS TO RUN
GPM002-F.....	PONY MOTOR #2 FAILS TO START
GPM002-G.....	PONY MOTOR #2 FAILS TO RUN
GPM003-F.....	PONY MOTOR #3 FAILS TO START
GPM003-G.....	PONY MOTOR #3 FAILS TO RUN
GPM004-F.....	PONY MOTOR #4 FAILS TO START
GPM004-G.....	PONY MOTOR #4 FAILS TO RUN
GPM005-F.....	PONY MOTOR #5 FAILS TO START
GPM005-G.....	PONY MOTOR #5 FAILS TO RUN
GPP24HDL.....	PRIMARY COOLING SYSTEM INLET HEADER PIPE BREAK
GPP26HDL.....	PRIMARY COOLING SYSTEM OUTLET HEADER PIPE BREAK
GREBAILP.....	FAILURE OF RECTIFIER/CHARGER
GRL379-M.....	RELAY COMPENSATOR REL-379 FAILS HIGH
GRL380-M.....	RELAY COMPENSATOR REL-380 FAILS HIGH
GRL384-M.....	RELAY COMPENSATOR REL-384 FAILS HIGH
GRL384BM.....	RELAY REL-384B FAILS HIGH
GRL8009H.....	RELAY CR-8009 FAILS OPEN CIRCUIT
GRL8010H.....	RELAY CR-8010 FAILS OPEN CIRCUIT
GSV2481A.....	INST AIR VALVE CASV-248-1 FTOD
GSV2511A.....	INST AIR VALVE CASV-251-1A FTOD
GSV2512A.....	INST AIR VALVE CASV-251-2A FTOD
GSW204-H.....	SELECTOR SWITCH SS-204 DEVELOPS OPEN CIRCUIT
GSW241-J.....	SWITCH 241, FAILS SHORT TO POWER
GSW241-Q.....	SWITCH SS-241, OPERATOR FAILS TO CHANGE
GSW245-J.....	SWITCH SS-245, FAILS SHORT TO POWER
GSW245-Q.....	SWITCH SS-245, OPERATOR FAILS TO CHANGE
GSW2H33Q.....	INADVERT TRIP TO ECS FROM FAILURE TO BYPASS 2H33 AFTER SCRAM
GSW5K65H.....	STATIC INVERTER SWITCH 5K65 FAILS 'MID' POSITION
GSW749AJ.....	SWITCH CS-749A SHORT CIRCUIT TO POWER
GSW749AV.....	SWITCH CS-749A FAILURE TO RESTORE TO CORRECT POSITION
GSW8009I.....	SWITCH CS-8009 FAILS SHORT TO GROUND
GSW8010I.....	SWITCH CS-8010 FAILS SHORT TO GROUND
GSWTRANH.....	FAILURE OF STATIC INVERTER TRANSFER SWITCH TO TRANSFER
GTE248-M.....	THERMOCOUPLE RTD-TT-248 FAILS HIGH
GTE250-M.....	THERMOCOUPLE RTD-TT-250 FAILS HIGH
GXR1165P.....	FAILURE OF 208/120 VAC TRANSFORMER 11X65
GXR7X65P.....	FAILURE OF 120/118 VAC VOLT REGULATING TRANSFORMER 7X65
HAV1051C.....	FPDV-105-1 FTRO, FW-1 FEEDWATER CONTROL VALVE
HAV1141C.....	FPDV-114-1 FTRO, FW-1 FEEDWATER CONTROL VALVE
HAV1241C.....	FPDV-124-1 FTRO, CE-1 FEEDWATER CONTROL VALVE
HAV1242C.....	FPDV-124-2 FTRO, CE-2 FEEDWATER CONTROL VALVE
HAVCV01E.....	TG STEAM CONTROL VALVE #1 FAILS CLOSED
HAVCV02E.....	TG STEAM CONTROL VALVE #2 FAILS CLOSED
HBRCE-1P.....	CE-1 OIL FIRED BOILER FAILURE
HBRCE-1X.....	CE-1 BOILER UNAVAILABLE DUE TO MAINT
HBRCE-2P.....	CE-2 OIL FIRED BOILER FAILURE
HBRCE-2X.....	CE-2 BOILER UNAVAILABLE DUE TO MAINTENANCE
HBRFW-1P.....	FW-1 OIL FIRED BOILER FAILURE
HBRFW-1X.....	FW-1 BOILER UNAVAILABLE DUE TO MAINTENANCE
HBU138AH.....	LOSS OF 13.8KV BUS A DUE TO HARDWARE OR PERIPHERAL FAILURES

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
HBU1388H.....	LOSS OF 13.8KV BUS B DUE TO HARDWARE OR PERIPHERAL FAILURES
HBU416AH.....	LOSS OF 4160 V BUS A DUE TO HARDWARE OR PERIPHERAL FAILURES
HBU4168H.....	LOSS OF 4160 V BUS B DUE TO HARDWARE OR PERIPHERAL FAILURES
HBU480AH.....	LOSS OF 480 V BUS A DUE TO HARDWARE OR PERIPHERAL FAILURES
HBU4808H.....	LOSS OF 480 V BUS B DUE TO HARDWARE OR PERIPHERAL FAILURES
HCLAIR-L.....	TG CONDENSER CASING AIR LEAKAGE (3E-6/HR)
HCLOILSL.....	TG OIL COOLER SHELL SIDE RUPTURE (3E-6/HR)
HCLOILTE.....	TG OIL COOL TUBES PLUGD (3E-5/HR) ASSUME 1 TUBE PLUGD FAILS
HCLOILT.....	TG OIL COOL TUBES RUPT (3E-9/TUBE-HR) ASSUME 1 TUBE FAILS HX
HCLTUBEL.....	TG COND TUBE FAILURES (3E-9/TUBE-HR) ASSUME 1 TUBE FAILS HX
HCMTHWMP.....	TG CONDENSATE HOTWELL LEVEL CONTROL INSTRUMENTATION
HCPP004G.....	MAIN OIL PUMP FAILS TO CONTINUE RUNNING
HCV1041C.....	HPV-104-1 FTRO, TG STEAM STOP CHECK VALVE
HCV1042C.....	HPV-104-2 FTRO, TG STEAM STOP CHECK VALVE
HCV1081A.....	MWV-108-1, 3/4-INCH CHECK VALVE FAILS TO OPEN DEMAND
HCV1082A.....	MWV-108-2, 3/4-INCH CHECK VALVE FAILS TO OPEN ON DEMAND
HCV1221C.....	FPOV-122-1 FTRO, CE-1 FEEDWATER CHECK VALVE
HCV1222C.....	FPOV-122-2 FTRO, CE-2 FEEDWATER CHECK VALVE
HCV1541C.....	HPV-154-1 FTRO, CE-1 STEAM STOP CHECK VALVE
HCV1551C.....	HPV-155-1 FTRO, CE-2 STEAM STOP CHECK VALVE
HCVCV04A.....	TG EMERG BEARING & SEAL PUMP CV FAILS TO OPEN ON DEMAND
HCVCV05C.....	CHECK VALVE FAILS TO REMAIN OPEN
HCVCV06A.....	TG BEARING AND SEAL OIL PUMP CV FAILS TO OPEN ON DEMAND
HCVCV07A.....	TG AUX OIL PUMP CHECK VALVE FAILS TO OPEN ON DEMAND
HCV1051C.....	CONDV-105-1, 6-INCH CONTROL VALVE FAILS TO REMAIN OPEN
HFV0101C.....	RSWV-101 FTRO, TG CRW SUPPLY VALVE
HGV1072C.....	MWV-107-2, 3/4-INCH MANUAL GLOBE VALVE FAILS TO REMAIN OPEN
HGV1073C.....	MWV-107-3, 3/4-INCH MANUAL GLOBE VALVE FAILS TO REMAIN OPEN
HGV1119C.....	SELF COOLING FAILS, CONDV-111-19 FAILS TO REMAIN OPEN
HGV1120C.....	SELF COOLING FAILS, CONDV-111-20 FAILS TO REMAIN OPEN
HMB0102C.....	RWRV-102 FTRO, TG CRW RETURN CONTROL VALVE
HMG1071C.....	FPOV-107-1 FTRO, FW-1 FEEDWATER CONTROL VALVE
HMG1251C.....	FPOV-125-1 FTRO, CE-1 FEEDWATER CONTROL VALVE
HMG1252C.....	FPOV-125-2 FTRO, CE-2 FEEDWATER CONTROL VALVE
HMG1431C.....	FPOV-143-1 FTRO, CE-1 FEEDWATER VALVE
HMG1432C.....	FPOV-143-2 FTRO, CE-2 FEEDWATER VALVE
HMP1388F.....	AUXILIARY OIL PUMP FAILS TO START
HMP1388G.....	AUXILIARY OIL PUMP FAILS TO CONTINUE RUNNING
HMP1388X.....	AUXILIARY L PUMP UNAVAILABLE DUE TO MAINTENANCE
HMPP003G.....	BEARING & SEAL OIL PUMP FAILS TO CONTINUE RUNNING
HMPP005F.....	EMERGENCY BEARING & SEAL PUMP MOTOR FAILS TO START
HMPP005G.....	EMERGENCY BEARING & SEAL OIL PUMP FAILS TO CONTINUE RUNNING
HMPP005X.....	EMERGENCY BEARING & SEAL OIL PUMP UNAVAILABLE DUE TO MAINT
HMPTGCAG.....	TG CONDENSATE PUMP A FAILS TO RUN
HMPTGCBG.....	TG CONDENSATE PUMP B FAILS TO RUN
HMPVP01G.....	TG CONDENSATE VACUUM PUMP #1 FAILS TO CONTINUE RUNNING
HMPVP02F.....	TG CONDENSATE VACUUM PUMP #2 FAILS TO START
HMPVP02G.....	TG CONDENSATE VACUUM PUMP #2 FAILS TO CONTINUE RUNNING
HMPVP02X.....	TG CONDENSATE VAC PUMP #2 UNAVAILABLE DUE TO MAINTENANCE
HPV1051C.....	HPV-105 FTRO, FW-1 STOP VALVE
HPV1071C.....	HPV-107-1 FTRO, STEAM CONTROL VALVE
HPV1081C.....	HPV-108-1 FTRO, STEAM CONTROL VALVE
HTG1388G.....	TG HARDWARE FAILURE (1 EVENT/(23YR*4400FP-HR/YR)=9.88E-6/HR)
HTG1388P.....	TG EXCITER FAILURE (.33/(23YR*4400FP-HR/YR)=3.26E-6/HR)
HTV1011C.....	CONDV-101-1, 8-INCH MANUAL GATE VALVE FAILS TO REMAIN OPEN
HTV1012C.....	CONDV-101-2, 8-INCH MANUAL GATE VALVE FAILS TO REMAIN OPEN

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
HTV1031C.....	HPV-103-1 FTRO, STEAM ISOLATION VALVE
HTV1041C.....	CONDV-104-1, 6-INCH MANUAL GATE VALVE FAILS TO REMAIN OPEN
HTV1042C.....	CONDV-104-2, 6-INCH MANUAL GATE VALVE FAILS TO REMAIN OPEN
HTV1043C.....	CONDV-104-3, 6-INCH MANUAL GATE VALVE FAILS TO REMAIN OPEN
HTV1044C.....	CONDV-104-4, 6-INCH MANUAL GATE VALVE FAILS TO REMAIN OPEN
HTV1201C.....	FPDV-120-1 FTRO, CE-1,2 FEEDWATER BLOCK VALVE
HTV1501C.....	HPV-150 FTRO, FW-1 STEAM ISOLATION VALVE
HTV1521C.....	HPV-152 FTRO, CE-1 STEAM ISOLATION VALVE
HTV1531C.....	HPV-153 FTRO, CE-2 STEAM ISOLATION VALVE
HTV2072C.....	HPV-207-2 24-INCH GATE VALVE FAILS TO REMAIN OPEN
HTV2073C.....	HPV-207-3 24-INCH GATE FAILS TO REMAIN OPEN
HZZBUSAP.....	BPA & 230KV HANF LOOP PWR FAIL (1/(23YR*4400FP-HR/YR=9.88E-6
IE-10.....	LOSS OF PCS PRESSURE CONTROL
IE-11.....	BAILEY CONTROL SYSTEM POWER FAILURE
IE-12A.....	TRANSIENT INITIATOR: GSCS DUMP - ONCE-THROUGH MODE
IE-12B.....	TRANSIENT INITIATOR: GSCS COMPLETE LOSS
IE-13A.....	SUPPORT SYSTEM INITIATOR: STATION BLACKOUT
IE-13B.....	SUPPORT SYSTEM INITIATOR: ELECTRICAL A BUS FAILURES
IE-13C.....	SUPPORT SYSTEM INITIATOR: ELECTRICAL B BUS FAILURES
IE-130-1.....	SUPPORT SYSTEM INITIATOR: ELECTRICAL DC FAILURES - J001
IE-130-2.....	SUPPORT SYSTEM INITIATOR: ELECTRICAL DC FAILURES - J050
IE-130-3.....	SUPPORT SYSTEM INITIATOR: ELEC. DC FAILURES - J200 & J200AC
IE-130-5.....	SUPPORT SYSTEM INITIATOR: ELECTRICAL DC FAILURES - J400
IE-130-6.....	SUPPORT SYSTEM INITIATOR: ELEC. DC FAILURES - J450 & J450AC
IE-13E-1.....	SUPPORT SYSTEM TRANSIENT: AIR SYSTEM FAILURES - N100
IE-13E-2.....	SUPPORT SYSTEM TRANSIENT: AIR SYSTEM FAILURES - N103
IE-13E-3.....	SUPPORT SYSTEM TRANSIENT: AIR SYSTEM FAILURES - N127
IE-1A.....	TRANSIENT INITIATOR: DRIVE TURBINE TRIP
IE-1B.....	TRANSIENT INITIATOR: LOSS OF FLOW TO A CELL
IE-2.....	TRANSIENT INITIATOR: LOSS OF HPI
IE-5.....	TRANSIENT INITIATOR: STEAM LINE BREAK (INSIDE CONFINEMENT)
IE-6.....	TRANSIENT INITIATOR: STEAM LINE BREAK (OUTSIDE CONFINEMENT)
IE-7A.....	TRANSIENT INITIATOR: FAILURE OF FDVP-6203 VALVES
IE-7B.....	TRANSIENT INITIATOR: FAILURE OF CONDV-243 VALVE
IE-8A.....	TRANSIENT INITIATOR: LINE BREAK IN COND I HEADER
IE-8B.....	TRANSIENT INITIATOR: FAILURE OF TWO CONDV-216S IN A CELL
IE-9.....	TRANSIENT INITIATOR: MAIN STEAM SYSTEM INITIATORS
INTFLOCA.....	FAILURE OF CV2R's & CV1's - INTRFC LOCA
JBC105AH.....	RECTIFIER FOR BATTERY A 105-N FAILS
JBC105BH.....	RECTIFIER FOR BATTERY B 105-N FAILS
JBC105DH.....	RECTIFIER D SERVING BATT. D & E FAILS
JBC109AH.....	RECTIFIER FOR BATTERY A 109-N FAILS
JBC109BH.....	RECTIFIER FOR BATTERY B 109-N FAILS
JBC153AH.....	RECTIFIER FOR BATTERY A 153-N FAILS
JBC153BH.....	RECTIFIER FOR BATTERY B 153-N FAILS
JBT105AK.....	LOSS OF PWR FROM 105-N BATTERY A
JBT105BK.....	LOSS OF PWR FROM 105-N BATTERY B
JBT105DK.....	LOSS OF PWR FROM 105-N BATTERY D
JBT105EK.....	LOSS OF PWR FROM 105-N BATTERY E
JBT109AK.....	LOSS OF PWR FROM 109-N BATTERY A
JBT109BK.....	LOSS OF PWR FROM 109-N BATTERY B
JBT153AK.....	LOSS OF PWR FROM 153-N BATTERY A
JBT153BK.....	LOSS OF PWR FROM 153-N BATTERY B
JBU105AH.....	125V DC BUS A FOR 105-N HARDWARE FAULTS
JBU105BH.....	125V DC BUS B FOR 105-N HARDWARE FAULTS
JBU105DH.....	24V DC BUS A FOR 105-N HARDWARE FAULTS

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
JBU105EH.....	24V DC BUS B FOR 105-N HARDWARE FAULTS
JBU109AH.....	125V DC BUS A FOR 109-N HARDWARE FAULTS
JBU109BH.....	125V DC BUS B FOR 109-N HARDWARE FAULTS
JBU153AH.....	125V DC BUS A FOR 153-N HARDWARE FAULTS
JBU153BH.....	125V DC BUS FOR 153-N HARDWARE FAULTS
JSW109AH.....	FAILURE OF SWITCHOVER TO 109 A DC POWER
JSW109BH.....	FAILURE OF SWITCHOVER TO 109 B DC POWER
K115MOD.....	CRW PUMP 1 SUCT GATES, TRAV SCRNS FAIL
K116MOD.....	CRW PUMP 1 MOTOR COOLING FAILS
K125MOD.....	CRW PUMP 2 SUCT GATES, TRAV SCRNS FAIL
K126MOD.....	CRW PUMP 2 MOTOR COOLING FAILS
K135MOD.....	CRW PUMP 3 SUCT GATES, TRAV SCRNS FAIL
K136MOD.....	CRW PUMP 3 MOTOR COOLING FAILS
K145MOD.....	CRW PUMP 4 SUCT GATES, TRAV SCRNS FAIL
K146MOD.....	CRW PUMP 4 MOTOR COOLING FAILS
K151MOD.....	CRW SCRNSH PUMP 1 FAILS TO RUN
K153MOD.....	CRW SCRNSH PUMP 2 FAILS TO RUN
K210MOD.....	RWS-1 PUMP 1 FAILS TO OPERATE
K220MOD.....	RWS-1 PUMP 2 FAILS TO OPERATE
K230MOD.....	RWS-1 PUMP 3 FAILS TO OPERATE
K310MOD.....	RWS-2 PUMP 1 FAILS TO OPERATE
K320MOD.....	RWS-2 PUMP 2 FAILS TO OPERATE
K330MOD.....	RWS-2 PUMP 3 FAILS TO OPERATE
K411MOD.....	FSDP 1 STARTING AIR SUPPLY FAILS
K413MOD.....	FSDP 1 B/U STARTING AIR FAILS (RECOVERY)
K420MOD.....	FSDP 1 FAILS ON LOSS OF FUEL OIL
K432AMOD.....	FSDP 1 EXPANSION TANK FILL CONTROL FAILURES
K434MOD.....	FSDP 1 HEAT EXCHANGER COOLING FAILS
K450MOD.....	FSDP 1 VALVES FAIL TO REMAIN OPEN
K461MOD.....	FSDP 2 STARTING AIR SUPPLY FAILS
K463MOD.....	FSDP 2 B/U STARTING AIR FAILS (RECOVERY)
K470MOD.....	FSDP 2 FAILS ON LOSS OF FUEL OIL
K482AMOD.....	FSDP 2 EXPANSION TANK FILL CONTROL FAILURES
K484MOD.....	FSDP 2 HEAT EXCHANGER COOLING FAILS
K490MOD.....	FSDP 2 VALVES FAIL TO REMAIN OPEN
K521MOD.....	LLDP SUPPLY LINES TO FSDP FAIL
K550MOD.....	FSDP COOLING WATER SUPPLY FAILS
KCBCRW3K.....	CRW PUMP 3 NOT RACKED IN
KCV1061A.....	SERV WATER PUMP 1 RWSV-106-1 FAILS TO OP
KCV1062A.....	SERV WATER PUMP 2 RWSV-106-2 FAILS TO OP
KCV5031A.....	CRW SCRNSH PMP 1 SWAV-503-1 FAILS TO OP
KCV5032A.....	CRW SCRNSH PMP 2 SWAV-503-2 FAILS TO OP
KCV5081A.....	CRW SCRNSH PMP 1 SWAV-508-1 FAILS TO OP
KCV5082A.....	CRW SCRNSH PMP 2 SWAV-508-2 FAILS TO OP
KCV5511C.....	CRW SUP HDR RWS-2 RWSV-551-1 FAILS TRO
KCV553FA.....	FW TANK TO FSDP RWSV-553-3 FAILS TO OPEN
KCV5653A.....	RWS-1 PMP 3 RWSV-565-3 FAILS TO OPEN
KCV5693A.....	RWS-2 PMP 3 RWSV-569-3 FAILS TO OPEN
KCV5861A.....	DW TANK TO FSDP RWSV-586-1 FAILS TO OPEN
KCV8061C.....	CRW PUMP 1 DISCH RWSV-806-1 FAILS TRO
KCV8062C.....	CRW PUMP 2 DISCH RWSV-806-2 FAILS TRO
KCV8063A.....	CRW PMP 3 DISCH RWSV-806-3 FAILS TO OPEN
KCV8063C.....	CRW PUMP 3 RWSV-806-3 FAILS TRO
KCV8064C.....	CRW PUMP 4 DISCH RWSV-806-4 FAILS TRO
KDPFSD1F.....	FSDP 1 FAILS TO START
KDPFSD1G.....	FSDP 1 FAILS TO OPERATE

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
KDPFSD2F.....	FSDP 2 FAILS TO START
KDPFSD2G.....	FSDP 2 FAILS TO OPERATE
KDUF11NE.....	FSDP 1 COMBUSTION AIR INTAKE HDR PLUGS
KDUF10UE.....	FSDP 1 EXHAUST AIR HEADER PLUGS
KDUF21NE.....	FSDP 2 COMBUSTION AIR INTAKE HDR PLUGS
KDUF20UE.....	FSDP 2 EXHAUST AIR HEADER PLUGS
KFV5522C.....	FSDP SUPPLY HDR RWSV-552-2 FAILS TRO
KFV5551C.....	FW TANK TO FSDP RWSV-555-1 FAILS TRO
KFV5552A.....	FW TANK TO FSDP RWSV-555-2 FAILS TO OPEN
KFV5572A.....	DW TANK TO FSDP RWSV-557-2 FAILS TO OPEN
KFV5671C.....	FSDP HEX SUP RWSV-567-1 FAILS TO REM OP
KGV5063C.....	CRW PUMP 3 PRE-LUBE RWSV-506-3 FAILS TRO
KHB8051C.....	CRW PUMP 1 DISCH RWSV-805-1 FAILS TRO
KHB8052C.....	CRW PUMP 2 DISCH RWSV-805-2 FAILS TRO
KHB8053A.....	CRW PUMP 3 DISCH RWSV-805-3 FAILS TO OP
KHB8053C.....	CRW PUMP 3 DISCH RWSV-805-3 FAILS TRO
KHB8054C.....	CRW PUMP 4 DISCH RWSV-805-4 FAILS TRO
KHOUSE1.....	STANDBY CRW PUMP AVAILABLE FOR RECOVERY
KHOUSE2.....	FSDP B/U WATER SUPPLIES AVAIL (RECOVERY)
KMB8011C.....	CRW SUP HDR RWSV-801-1 FAILS TO REM OPEN
KMB8012C.....	CRW SUP HDR RWSV-801-2 FAILS TO REM OPEN
KMB8021C.....	CRW SUP HDR RWSV-802-1 FAILS TO REM OPEN
KMB8022C.....	CRW SUP HDR RWSV-802-2 FAILS TO REM OPEN
KMPCRW1G.....	CRW PUMP 1 FAILS TO OPERATE
KMPCRW2G.....	CRW PUMP 2 FAILS TO OPERATE
KMPCRW3F.....	CRW PUMP 3 FAILS TO START WHEN REQD
KMPCRW3G.....	CRW PUMP 3 FAILS TO OPERATE
KMPCRW3W.....	CRW PUMP 3 OUT FOR TESTING
KMPCRW3X.....	CRW PUMP 3 OUT FOR MAINTENANCE
KMPCRW4G.....	CRW PUMP 4 FAILS TO OPERATE
KMPHRP3F.....	RWS-2 PUMP 3 FAILS TO START
KMPHRP3W.....	RWS-2 PUMP 3 OUT FOR TESTING
KMPHRP3X.....	RWS-2 PUMP 3 OUT FOR MAINTENANCE
KMPLRP3F.....	RWS-1 PUMP 3 FAILS TO START
KMPLRP3W.....	RWS-1 PUMP 3 OUT FOR TESTING
KMPLRP3X.....	RWS-1 PUMP 3 OUT FOR MAINTENANCE
KMPSER1F.....	SERV WATER PUMP 1 FAILS TO START
KMPSER1G.....	SERV WATER PUMP 1 FAILS TO CONTINUE RUN
KMPSER2F.....	SERV WATER PUMP 2 FAILS TO START
KMPSER2G.....	SERV WATER PUMP 2 FAILS TO CONTINUE RUN
KMPSWP1F.....	CRW SCRNVSH PUMP 1 FAILS TO START
KMPSWP2F.....	CRW SCRNVSH PUMP 2 FAILS TO START
KOP5552R.....	OPER FAILS TO OPEN FW TANK VALVE
KOP5572R.....	OPER FAILS TO OPEN DW TANK VALVE
KOPCRW3R.....	OPER FAILS TO START CRW PUMP 3 WHEN REQD
KOPFSD1V.....	FSDP 1 START SIGNAL BYPASSED
KOPFSD2V.....	FSDP 2 START SIGNAL BYPASSED
KOPFSDPQ.....	OPER FAILS TO START FSDPS
KOPHRP3Q.....	LOCAL OPER FAILS TO START RWS-2 PUMP 3
KOPLRP3Q.....	LOCAL OPER FAILS TO START RWS-1 PUMP 3
KOPSBUSQ.....	OPER FAILS TO SWITCH TO POWERED BUS
KOPSERWQ.....	OPER FAILS TO START SERV WATER PUMPS
KOPSWPSQ.....	OPER FAILS TO START SCREENWASH PUMPS
KPP1020E.....	102" OUTFALL PLUGS
KPP1080E.....	RWR 108-IN HEADER PLUGS
KPP6600E.....	RWR 66-IN HEADER PLUGS

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
KPPCRWSL.....	CRW HEADERS FAIL
KPPEMERE.....	EMERGENCY OUTFALL PLUGS
KPPFSDPL.....	FSDP PIPING FAILS
KPPLRDHL.....	RWS-1 HEADERS FAIL
KPPRWS2L.....	RWS-2 HEADERS FAIL
KPPSRWRL.....	SERVICE WATER HEADERS FAIL
KPS116XM.....	SERV WATER PMPS PRESS SWIT PS-116 FAILS
KPS58400.....	FSDP START SIGNAL PS-584 FAILS
KPS584A0.....	FSDP START SIGNAL PS-584A FAILS
KSCREEN.....	CRW SCREENWASH PUMPS REQUIRED FOR
KSL0001E.....	SEALWELL 1 PLUGS
KSL0002E.....	SEALWELL 2 PLUGS
KSW116X0.....	SERV WATER PUMPS START SWITCH FAILS
KSWSWP10.....	CRW SCRWSH DP CONT #1 START SIGNAL FAILS
KSWSWP20.....	CRW SCRWSH DP CONT #2 START SIGNAL FAILS
KTKFS1AL.....	FSDP 1 START AIR ACCUMULATOR FAILS
KTKFS1XL.....	FSDP 1 EXPANSION TANK EMPTY
KTKFS2AL.....	FSDP 2 START AIR ACCUMULATOR FAILS
KTKFS2XL.....	FSDP 2 EXPANSION TANK EMPTY
KTT5524A.....	FSDP 1 JACKET CLG JWV-552-4 FAILS TO OP
KTT5525A.....	FSDP 2 JACKET CLG JWV-552-5 FAILS TO OP
KTV1021C.....	SERV WATER SUP HDR RWSV-102-1 FAILS TRO
KTV1051C.....	SERV WATER PUMP 1 RWSV-105-1 FAILS TRO
KTV1052C.....	SERV WATER PUMP 2 RWSV-105-2 FAILS TRO
KTV1053C.....	SERV WATER PUMP 1 RWSV-105-3 FAILS TRO
KTV1054C.....	SERV WATER PUMP 2 RWSV-105-4 FAILS TRO
KTV5081C.....	CRW PMP 3 PR-LUB CLG RWV-508-1 FAILS TRO
KTV5511C.....	FSDP 1 START AIR CAEV-551-1 FAILS TRO
KTV5512C.....	FSDP 2 START AIR CAEV-551-2 FAILS TRO
KTV599XC.....	CRW SUP HDR RWS-2 RWSV-599 FAILS TRO
KTV8031C.....	FSDP 1 FUEL OIL DOSV-803-1 FAILS TRO
KTV8172C.....	FSDP 2 FUEL OIL DOSV-817-2 FAILS TRO
KZZHYD2P.....	HYD POWERPACK #2 FAILS
L253MOD.....	LPFWP 3 FAILS TO START
L254MOD.....	LPFWP 4 FAILS TO START
L261MOD.....	LPFWP 1 FAILS TO DELIVER FLOW
L262MOD.....	LPFWP 2 FAILS TO DELIVER FLOW
L263MOD.....	LPFWP 3 FAILS TO DELIVER FLOW
L264MOD.....	LPFWP 4 FAILS TO DELIVER FLOW
LCV5581D.....	LPFW RWSV-558-1 FAILS TO REMAIN CLOSED
LFV5511C.....	FWV-551-1 FAILS TO REMAIN OPEN
LFV5512C.....	FWV-551-2 FAILS TO REMAIN OPEN
LFV8112C.....	FWV-811-2 FAILS TO REMAIN OPEN
LLOCA.....	LARGE LOSS OF COOLANT ACCIDENT > 2.7-INCH DIAMETER HOLE
LMPLFP4F.....	LPFWP 4 FAILS TO START
LOPSWITQ.....	OPER FAILS TO SWITCH TO POWERED BUS
LPPFWS1L.....	LPFW SYSTEM HEADERS FAIL
LTKFWTOL.....	FWT LESS THAN 25% FULL
LTV5591D.....	LPFW RWSV-559-1 FAILS TO REMAIN CLOSED
M101MOD.....	DW TANK SUPPLY FAILS
M110MOD.....	LPI PUMP 1 FAILS TO DELIVER FLOW
M120MOD.....	LPI PUMP 2 FAILS TO DELIVER FLOW
M130MOD.....	LPI PUMP 3 FAILS TO DELIVER FLOW
M160MOD.....	LPI PUMP 3 FAILS TO RUN
M202MOD.....	AHR TANK SUPPLY TO AHR PUMPS FAILS
M210MOD.....	AHR PUMP 1 FAILS TO DELIVER FLOW

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
M211MOD.....	AHR PUMP 1 SUPPLY FAILS
M220MOD.....	AHR PUMP 2 FAILS TO DELIVER FLOW
M221MOD.....	AHR PUMP 2 SUPPLY FAILS
M230MOD.....	AHR PUMP 3 FAILS TO DELIVER FLOW
M240MOD.....	AHR PUMP 1 FAILS TO START WHEN REQD
M250MOD.....	AHR PUMP 2 FAILS TO START WHEN REQD
M260MOD.....	AHR PUMP 3 FAILS TO START WHEN REQD
M310MOD.....	SLMU PUMP 1 FAILS TO DELIVER FLOW
M320MOD.....	SLMU PUMP 2 FAILS TO DELIVER FLOW
MCV5641C.....	LPI DISCH HDR DWV-564-1 FAILS TO REM CL
MDV5621C.....	LPI DISCH HDR DWV-562-1 FAILS TO REM OP
MDV562XC.....	AHR DISCH HDR MWV-562-1 FAILS TO REM OP
MFV8053C.....	AHR TANK SUP DWV-805-3 FAILS TO REM OPEN
MLSSTLVD.....	LOW ST LVL SWIT FAILS TO START AHR PMPS
MMPLPI3F.....	LPI PUMP 3 FAILS TO START
MOPAHDAV.....	AHRP DISCONTINUOUS ACTION TRIP BYPASSED
MOPAHRR.....	OPER FAILS TO START AHR PUMPS
MOPLBUSQ.....	OPER FAILS TO SWITCH TO POWERED BUS FOR LPI PUMP #3
MOPLP13Q.....	OPER FAILS TO START LPI PUMP 3 WHEN REQD
MOPSBUSQ.....	OPER FAILS TO SWITCH TO POWERED BUS FOR AHR PUMP #3
MPPAHRSL.....	AHR PIPING FAILS
MPPLPISL.....	LPI PIPING FAILS
MPPSLMUL.....	SLMU PIPING FAILS
MTKAHRTL.....	AHR TANK LESS THAN 25% FULL
MTKDWTL.....	INSUFFICIENT WATER IN DW TANK
N100AMOD.....	I.A. TO 184-N ANNEX FAILS
N107MOD.....	LOSS OF AIR DRYER #2 TRAIN
N108MOD.....	LOSS OF SERV AIR COOL/SEP
N108XMOD.....	LOSS OF SERV AIR SEP/COOLER
N109MOD.....	VALVES CASV-186-1,2 FAIL TO REMAIN OPEN
N111MOD.....	LOSS OF AIR TO CASV-1100-3
N123MOD.....	LOSS OF BACKUP IAIR VIA SERV AIR X-TIE
N127BMOD.....	LOSS OF LPFW COOLING WATER
N127CMOD.....	LOSS OF SERVICE WATER BACKUP COOLING
N131AMOD.....	LOSS OF BREATHING AIR SUPPLY (1)
N131BMOD.....	LOSS OF BREATHING AIR SUPPLY (2)
N131MOD.....	LOSS OF BREATHING AIR SUPPLY (3)
NAG0185A.....	VALVE CASV-185 FAIL TO OPEN ON DEMAND
NAG1031E.....	SAV-103-1 FAILS TO REMAIN OPEN PLUG
NAG1061A.....	VALVE CASV-106-1 FAILS TO OPEN ON DEMAND
NAG1081A.....	CASV-108-1 FAILS TO OPEN ON DEMAND
NCRA1CLP.....	COMP #1 AIR COOLING FAILS
NCRA2CLP.....	BOOSTER COMP #2 AIR COOLING FAILS
NCRCOM1F.....	BOOSTER COMP #1 FAILS TO START
NCRCOM1G.....	BOOSTER COMP #1 FAILS TO RUN
NCRCOM2F.....	BOOSTER COMP #2 FAILS TO START
NCRCOM2G.....	BOOSTER COMP #2 FAILS TO RUN
NCV1071C.....	VALVE CASV 107-1 FAILS TO REMAIN OPEN
NCVCVA1E.....	CV-A-1 FAILS TO REMAIN OPEN PLUG
NCVCVB1E.....	CV-B-1 FAILS TO REMAIN OPEN PLUG
NDYCOM1E.....	ECS AIR DRYER #1 PLUGS
NDYCOM2E.....	ECS AIR DRYER #2 PLUGS
NDYDOT1E.....	ECS AIR DRYER #1 OUTLET VALVE FAILS PLUG
NDYDOT2E.....	ECS AIR DRYER #2 OUTLET VALVE FAILS PLUG
NFLCOM1E.....	COMP #1 INLET AIR FILTER PLUGS
NFLCOM2E.....	COMP #2 INLET AIR FILTER PLUGS

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
NGV1003E.....	CASV-1100-3 FAILS TO REMAIN OPEN
MOTLLOCA.....	FLAG TO INDICATE EITHER SMALL LOCA OR TRANSIENT
NPPAIRSL.....	COMPRESSED AIR PIPING FAILS
NPS0001U.....	MISCALIBRATION: IAC #1 PRESSURE SWITCHES
NPS0002U.....	MISCALIBRATION: IAC #2 PRESSURE SWITCHES
NPS0107N.....	PS-107 FAILS LOW-CLOSING SAV-103-1
NPS0107U.....	MISCALIBRATION: PS-107
NPSSAC1U.....	MISCALIBRATION: SAC #1 PRESSURE SWITCHES
NPV5756E.....	CASV-157-56 FAILS TO REMAIN OPEN
NPVAREGE.....	A BANK REG FAILS TO REMAIN OPEN PLUG
NPVAREGU.....	A BANK REGULATOR MISCALIBRATED
NPVBREGE.....	B BANK REG FAILS TO REMAIN OPEN PLUG
NPVBREGU.....	B BANK REGULATOR MISCALIBRATED
NRV3181D.....	IAV-3318-1 FAILS TO REMAIN CLOSED
NRV3182D.....	IAV-3318-2 FAILS TO REMAIN CLOSED
NRV3289D.....	IAV-3289 FAILS TO REMAIN CLOSED
NSV1531E.....	CASV-153-1 FAILS TO REMAIN OPEN
NTKBOTAL.....	6 A BANK BOTTLES LEAK
NTKBOTBL.....	6 B BANK BOTTLES LEAK
NTV0310E.....	CASV-103-10 FAILS TO REMAIN OPEN PLUG
NTV0311C.....	VALVE CASV-103-11 FAILS TO REMAIN OPEN
NTV1045C.....	VALVE SAV-104-5 FAILS TO REMAIN OPEN
NTV1051C.....	VALVE CASV-105-1 FAILS TO REMAIN OPEN
NTV1052C.....	VALVE CASV-105-2 FAILS TO REMAIN OPEN
NTV1252D.....	VALVE CASV-125-2 FAILS TO REMAIN CLOSED
NTV1254D.....	VALVE CASV-125-4 FAILS TO REMAIN CLOSED
NTV5192E.....	CASV-151-92 FAILS TO REMAIN OPEN
NTV5193E.....	CASV-151-93 FAILS TO REMAIN OPEN
NTVABDVD.....	A BANK BLOWDOWN VALVE FAILS TO REM CLSED
NTVAVNTD.....	A BANK VENT FAILS TO REMAIN CLOSED
NTVBBDVJ.....	B BANK BLOWDOWN VALVE FAILS TO REM CLSED
NTVBVA1E.....	BV-A1 FAILS TO REMAIN OPEN PLUG
NTVBVA2E.....	BV-A2 FAILS TO REMAIN OPEN PLUG
NTVBVB1E.....	BV-B1 FAILS TO REMAIN OPEN PLUG
NTVBVB2E.....	BV-B2 FAILS TO REMAIN OPEN PLUG
NTVBVNTD.....	B BANK VENT FAILS TO REMAIN CLOSED
NTVC10TE.....	COMP #1 OUTLET VALVE FAILS TRO PLUG
NTVC20TE.....	COMP #2 OUTLET VALVE FAILS TRO PLUG
NTVNV11E.....	NV-1-1 FAILS TO REMAIN OPEN PLUG
NTVNV12E.....	NV-1-2 FAILS TO REMAIN OPEN PLUG
SLOCA.....	SMALL LOSS OF COOLANT ACCIDENT < 2.7-INCH
T101MOD.....	REACTOR GAS SYSTEM FAILURE
T11ER.....	OPER FAILS TO RECOVER AIR COMP LOSS DUE TO SINGLE A BUS FAIL
T13ER.....	OPERATOR FAILS TO RECOVER AIR FOR ALL PRI/SEC AIR LOSS FAILS
T2H33R1.....	OPERATOR FAILS TO RECOVER FROM GSW2H33Q FAILURE
T2H33R3.....	OPERATOR FAILS TO RECOVER AOP2H33V
THOUSE.....	TRANSIENT INITIATOR EVENT OCCURS
V003MOD.....	NO AIR FROM DRYER 1
V009MOD.....	DRYER 2 PATH FAILS
V010MOD.....	NO AIR FROM DRYER 2
V102MOD.....	DRYER 1 PATH FAILS
V3R.....	OPERATOR FAILS TO RECOVER V-3 FAILURES
VCV5751C.....	VALVE CAEV 575-1 FAILS TO REMAIN OPEN
VCV5752C.....	VLV CAEV 575-2 FAILS TO REMAIN OPEN
VCV5753C.....	VLV CAEV 575-3 FAILS TO REMAIN OPEN
VG5571E.....	VLV CAEV 557-1 FAILS TO REMAIN OPEN PLUG

Basic Event Descriptions (cont'd)

B.E. names	Descriptions:
VGV5572E.....	VLV CAEV 557-2 FAILS TO REMAIN OPEN PLUG
VGV5573E.....	VLV CAEV 557-3 FAILS TO REMAIN OPEN
VGV5861E.....	VLV CAEV-586-1 FAILS TO REMAIN OPEN PLUG
VGV5861V.....	FAIL TO REOPEN VLV CAEV-586-1 AFTER TEST
VGV5862E.....	VLV CAEV-586-2 FAILS TO REMAIN OPEN PLUG
VGV5862V.....	FAIL TO REOPEN VLV CAEV-586-2 AFTER TEST
VRCSAR1L.....	AIR RECEIVER 1 LEAKS
VRCSAR2L.....	AIR RECEIVER 2 LEAKS
VRCSAR3L.....	AIR RECEIVER 3 LEAKS
X003MOD.....	START CIRCUIT COMPONENTS FOR HLDP #1
X015MOD.....	START COMPONENTS FOR HLDP # 2
X023MOD.....	START COMPONENTS FOR HLDP #3
X030MOD.....	INDEPENDENT COMP. FOR LLDP #1
X038MOD.....	START COMP. FOR LLDP #2
X081MOD.....	START COMP. FOR LLDP #3
X100MOD.....	INDEPENDENT COMP. OF MASTER CIRCUIT ONE
X200MOD.....	INDEPEND. COMP. OF MASTER CIRCUIT TWO
XOP3H33V.....	OPER FAILS TO RESTORE 3H33 AFTER STARTUP
XOPHLD1Q.....	OPER. FAILS TO START #1 HLD FROM 105 CR
XOPHLD2Q.....	OPER. FAILS TO START #2 HLD FROM 105 CR
XOPHLD3Q.....	OPER. FAILS TO START #3 HLD FROM 105 CR
XOPLLD1Q.....	OPER. FAILS TO START #1 LLD FROM 105 CR
XOPLLD2Q.....	OPER. FAILS TO START #2 LLD FROM 105 CR
XOPLLD3Q.....	OPER. FAILS TO START #3 LLD FROM 105 CR
XRL1K33H.....	RELAY 1K33 FAIL TO ENGZ & CLOSE CONTACTS
XRL2K33H.....	RELAY 2K33 FAIL TO ENGZ & CLOSE CONTACTS
XRL793AH.....	RELAY 179K33A FAILS TO ENERGIZE
XRL793BH.....	RELAY 179K33B FAILS TO ENERGIZE
XRL793CH.....	RELAY 179K33C FAILS TO ENERGIZE
XRL793DH.....	RELAY 179K33D FAILS TO ENERGIZE
XRL793EH.....	RELAY 179K33E FAILS TO ENERGIZE
XRL793FH.....	RELAY 179K33F FAILS TO ENERGIZE
XRL803AH.....	RELAY 180K33A FAILS TO ENERGIZE
XRL803BH.....	RELAY 180K33B FAILS TO ENERGIZE
XRL803CH.....	RELAY 180K33C FAILS TO ENERGIZE
XRL803DH.....	RELAY 180K33D FAILS TO ENERGIZE
XRL803EH.....	RELAY 180K33E FAILS TO ENERGIZE
XRL803FH.....	RELAY 180K33F FAILS TO ENERGIZE
XRL88K3H.....	RELAY 88K33 FAIL TO ENGZ & CLOSE CONTACT
XRL89K3H.....	RELAY 89K33 FAIL TO ENGZ & CLOSE CONTACT
XRTS.....	FAILURE OF RTS TO SEND SIGNAL TO ESFAS

*****N REACTOR INITIATING EVENTS*****

C RVR EVENT
 $IEV(1) = BLDG-105 + PRIM-SHIELD + CORE-INLET-PIPES$
 $IE(1) = IEV(1)$

C LLOCA
 $IEV(2) = LLOCA-PIPES + BLDG-109$
 $IE(2) = IEV(2) * (1.0 - IEV(1))$

C SLOCA
 $IEV(3) = SLOCA-PIPES$
 $IE(3) = IEV(3) * ((1. - IEV(2)) * (1. - IEV(1)))$

C BLDG 182/109NT FAILURE
 $IEV(4) = BLDG-182 + BLDG-109NT$
 $IE(4) = IEV(4) * ((1. - IEV(3)) * (1. - IEV(2)) * (1. - IEV(1)))$

C T1 LOSP
 $A1 = HZZBUSAP + HBU138AH + HBU416AH + BLDG-151$
 $ABSTR = BLDG-153 + BLDG-184$
 $PLOSP = 1.0 - (1. - A1) * (1. - ABSTR)$

C
 $IEV(5) = PLOSP$
 $IE(5) = IEV(5) * ((1. - IEV(4)) * (1. - IEV(3)) * (1. - IEV(2)) * (1. - A002MOD))$

C T3 GENERAL TRANSIENT
 $IE(6) = 1.0 - (IE(10) + IE(2) + IE(3) + IE(4) + IE(5))$

BOOLEAN EXPRESSIONS WITH DOMINANT CUTSETS

C BOOL 1 = A*C (L7 AND T9-LOSP)

```

BOOL(1) =
KSL0001E * CCV3032B * ATKECCSL * ASLEVEL3 +
KSL0001E * CCV3031B * ATKECCSL * ASLEVEL3 +
KSL0002E * CCV3032B * ATKECCSL * ASLEVEL3 +
KSL0002E * CCV3031B * ATKECCSL * ASLEVEL3 +
CCV3032B * ATKECCSL * MTKDWTOL * ASLEVEL3 +
CCV3031B * ATKECCSL * MTKDWTOL * ASLEVEL3 +
KSL0002E * CDT0011A * CDT0012A * ATKECCSL * ASLEVEL3 +
KSL0001E * CDT0011A * CDT0012A * ATKECCSL * ASLEVEL3 +
KSL0002E * CDB3161A * CDB3162A * ATKECCSL * ASLEVEL3 +
KSL0001E * CDB3161A * CDB3162A * ATKECCSL * ASLEVEL3 +
CDT0011A * CDT0012A * ATKECCSL * MTKDWTOL * ASLEVEL3 +
CDB3161A * CDB3162A * ATKECCSL * MTKDWTOL * ASLEVEL3 +
COPPPMP3Q * CCV3032B * ATKECCSL * ASLEVEL3 +
MOPLPI3Q * CCV3032B * ATKECCSL * ASLEVEL3 +
CCV3031B * ATKECCSL * ASLEVEL3 * MOPLBUSQ +
COPSBUSQ * CCV3031B * ATKECCSL * ASLEVEL3 +
CCV3032B * ATKECCSL * ASLEVEL3 * MOPLBUSQ +
COPSBUSQ * CCV3032B * ATKECCSL * ASLEVEL3 +
COPPPMP3Q * CCV3031B * ATKECCSL * ASLEVEL3 +
MOPLPI3Q * CCV3031B * ATKECCSL * ASLEVEL3

```

C BOOL 2 = G*A*C (S9,T9-MISC)

C

C BOOL 2 CUTSETS W/O LOSP

```

BOOL(2) =
KSL0002E * CCV3032B * ATKECCSL * ASLEVEL3 * SLOCA +
KSL0002E * CCV3031B * ATKECCSL * ASLEVEL3 * SLOCA +
KSL0001E * CCV3031B * ATKECCSL * ASLEVEL3 * SLOCA +
KSL0001E * CCV3032B * ATKECCSL * ASLEVEL3 * SLOCA +
CCV3032B * ATKECCSL * MTKDWTOL * ASLEVEL3 +
CCV3031B * ATKECCSL * MTKDWTOL * ASLEVEL3 +
CDT0011A * CDT0012A * ATKECCSL * MTKDWTOL * ASLEVEL3 +
CDB3161A * CDB3162A * ATKECCSL * MTKDWTOL * ASLEVEL3

```

C LOSP TERMS IN BOOL 2

```

BOOL(11) =
COPPPMP3Q * CCV3032B * ATKECCSL * PLOSP * ASLEVEL3 +
MOPLPI3Q * CCV3032B * ATKECCSL * PLOSP * ASLEVEL3 +
KSL0001E * CCV3032B * ATKECCSL * PLOSP * ASLEVEL3 +
CCV3031B * ATKECCSL * PLOSP * ASLEVEL3 * MOPLBUSQ +
COPSBUSQ * CCV3031B * ATKECCSL * PLOSP * ASLEVEL3 +
KSL0002E * CCV3031B * ATKECCSL * PLOSP * ASLEVEL3 +
CCV3032B * ATKECCSL * PLOSP * ASLEVEL3 * MOPLBUSQ +
COPSBUSQ * CCV3032B * ATKECCSL * PLOSP * ASLEVEL3 +
KSL0002E * CCV3032B * ATKECCSL * PLOSP * ASLEVEL3 +
COPPPMP3Q * CCV3031B * ATKECCSL * PLOSP * ASLEVEL3 +
MOPLPI3Q * CCV3031B * ATKECCSL * PLOSP * ASLEVEL3 +
KSL0001E * CCV3031B * ATKECCSL * PLOSP * ASLEVEL3

```


C BOOL 3 = A*(-C) (L6 AND T8-LOSP)

BOOL(3) =
ATKECCSL * ASLEVEL3 +
ADPHL1-G * ADPHL3-G +
ADPHL1-G * ADPHL2-G +
ADPHL2-G * ADPHL3-G +
ADPHL3-G * XRLRA1AJ +
ADPHL2-G * XRLRA1AJ +
ADPHL1-G * XRLRA1BJ +
ADPHL3-G * XRLRA1BJ +
ADPHL1-G * XRLRA1CJ +
ADPHL2-G * XRLRA1CJ +
XRL13K3H * XRL9033H +
XRL2K33H * XRL9033H +
XRL89K3H * XRL13K3H +
XRL2K33H * XRL89K3H +
XRL88K3H * XRL13K3H +
XRL2K33H * XRL88K3H +
XRL1K33H * XRL89K3H +
XRL1K33H * XRL9033H +
XRL1K33H * XRL88K3H +
XRLRA1AJ * XRLRA1CJ +
XRLRA1EJ * XRLRA1FJ * ASLEVEL3 +
XRLRA1DJ * XRLRA1EJ * ASLEVEL3 +
XRLRA1BJ * XRLRA1CJ +
XRLRA1AJ * XRLRA1BJ +
XRLRA1DJ * XRLRA1FJ * ASLEVEL3 +
XOP3H33V +
AHOUSE1 * AOP2H33V * T2H33R3

C BOOL 4 = G*A*(-C) (S8,T8-MISC)
C (NO SLOCA OR LOSEP CUTSETS)
C

```

BOOL(4) =
ATKECCSL * ASLEVEL3 * HBU480AH +
KSL0002E * ATKECCSL * ASLEVEL3 * GHOUSE5 * ESCTURBP +
KSL0001E * ATKECCSL * ASLEVEL3 * GHOUSE5 * ESCTURBP +
KSL0002E * ATKECCSL * ASLEVEL3 * GHOUSE4 * ESCTURBP +
KSL0001E * ATKECCSL * ASLEVEL3 * GHOUSE4 * ESCTURBP +
ATKECCSL * MTKDWTOL * ASLEVEL3 +
KSL0001E * ATKECCSL * KTNCRP1F * ASLEVEL3 +
KSL0002E * ATKECCSL * KTNCRP1F * ASLEVEL3 +
KSL0001E * ATKECCSL * KTNCRP1G * ASLEVEL3 +
KSL0002E * ATKECCSL * KTNCRP1G * ASLEVEL3 +
KSL0002E * KMPSWP1F * ATKECCSL * ASLEVEL3 * KSCREEN +
KSL0001E * KMPSWP1F * ATKECCSL * ASLEVEL3 * KSCREEN +
ATKECCSL * KTNCRP1F * KTNCRP2F * KTNCRP4F * ASLEVEL3 +
ATKECCSL * KTNCRP1G * KTNCRP2F * KTNCRP4F * ASLEVEL3 +
ATKECCSL * KTNCRP1F * KTNCRP2F * KTNCRP4G * ASLEVEL3 +
ATKECCSL * KTNCRP1F * KTNCRP2G * KTNCRP4F * ASLEVEL3 +
ATKECCSL * KTNCRP1G * KTNCRP2G * KTNCRP4F * ASLEVEL3 +
ATKECCSL * KTNCRP1G * KTNCRP2F * KTNCRP4G * ASLEVEL3 +
ATKECCSL * KTNCRP1F * KTNCRP2G * KTNCRP4G * ASLEVEL3 +
ATKECCSL * KTNCRP1G * KTNCRP2G * KTNCRP4G * ASLEVEL3 +
KPP1080E * ATKECCSL * ASLEVEL3 * GHOUSE4 * ESCTURBP +
KPP6600E * ATKECCSL * ASLEVEL3 * GHOUSE4 * ESCTURBP +
KMPSWP1F * ATKECCSL * ASLEVEL3 * KSCREEN * KMPSWP2F +
ATKECCSL * KTNCRP1F * ASLEVEL3 * HTG138BP +
ATKECCSL * KTNCRP1G * ASLEVEL3 * HTG138BP +
ATKECCSL * LTKFWTOL * ASLEVEL3 * KMPSER1F * KMPSER2F +
ATKECCSL * KTNCRP1F * ASLEVEL3 * ETKDESTL +
ATKECCSL * KTNCRP1G * ASLEVEL3 * ETKDESTL +
KMPSWP1F * ATKECCSL * ASLEVEL3 * KSCREEN * HTG138BP +
ATKECCSL * KTNCRP1F * ASLEVEL3 * HCLOILTE +
ATKECCSL * KTNCRP1G * ASLEVEL3 * HCLOILTE +
KSL0002E * ATKECCSL * NFLTR11E * ASLEVEL3 +
KSL0001E * ATKECCSL * NOILASAP * ASLEVEL3 +
KSL0002E * ATKECCSL * NOILASAP * ASLEVEL3 +
KSL0001E * ATKECCSL * NFLTR11E * ASLEVEL3 +
ATKECCSL * KTNCRP1F * ASLEVEL3 * HCLOILSL +
ATKECCSL * KTNCRP1G * ASLEVEL3 * HCLOILSL +
ATKECCSL * KTNCRP1F * ASLEVEL3 * HCLOILTL +
HCLTUBEL * ATKECCSL * KTNCRP1F * ASLEVEL3 +
ATKECCSL * KTNCRP1G * ASLEVEL3 * HCLOILTL

```

C BOOL(5) = BOOL(4) CUTSETS WITH SLOCA TERM

```
BOOL(5) =
ATKECCSL * ASLEVEL3 * HBU416BH * SLOCA +
ATKECCSL * ASLEVEL3 * HBU138BH * SLOCA +
KSL0001E * ATKECCSL * ASLEVEL3 * SLOCA +
KSL0002E * ATKECCSL * ASLEVEL3 * SLOCA +
KPP6600E * ATKECCSL * ASLEVEL3 * SLOCA +
ATKECCSL * ASLEVEL3 * HTG138BP * SLOCA +
ATKECCSL * ASLEVEL3 * ETKDESTL * SLOCA +
ATKECCSL * ASLEVEL3 * HCLOILTE * SLOCA +
ATKECCSL * ASLEVEL3 * HCLOILSL * SLOCA +
ATKECCSL * ASLEVEL3 * HCLOILTL * SLOCA +
HCLTUBEL * ATKECCSL * ASLEVEL3 * SLOCA
```

C BOOL(6) = BOOL(4) CUTSETS WITH LOSEP, A BUSSES, BLDGS 151, 153, 184
C THESE ARE USED ONLY IN SLOCA SEQ'S

```
BOOL(6) =
ATKECCSL * PLOSP * ASLEVEL3 +
ADPHL1-G * ADPHL3-G * PLOSP +
ADPHL2-G * ADPHL3-G * PLOSP +
ADPHL1-G * ADPHL2-G * PLOSP
```

C BOOL 7 = A2 * G * (-A) * (-C) (S4, T4-MISC)

```
BOOL(7) =
KSL0001E * SLOCA * A2 +
KSL0001E * GHOUSE5 * ESCTURBP * A2 +
KSL0001E * GHOUSE4 * ESCTURBP * A2 +
KSL0002E * SLOCA * A2 +
KOPCRW3R * KHOUSE1 * A2 +
KSL0002E * GHOUSE5 * ESCTURBP * A2 +
KSL0002E * GHOUSE4 * ESCTURBP * A2 +
MTKDWTOL * A2 +
KSL0001E * KTNCRP1F * A2 +
KTNCRP3F * KHOUSE1 * A2 +
KSL0002E * KTNCRP1F * A2 +
KSL0001E * KTNCRP1G * A2 +
KTNCRP3G * KHOUSE1 * A2 +
KSL0002E * KTNCRP1G * A2
```

C
CBAR = 1.0
ABAR = 1.0
A2 = 2.73E-3

*****N REACTOR ACCIDENT SEQUENCES*****

C RVR EVENT
 ACC(1) = IE(1)

C LLOCA-6 = LLOCA*A*(-C)
 ACC(2) = IE(2)*BOOL(3)*CBAR

C LLOCA-7 = LLOCA*A*C
 ACC(3) = IE(2)*(BOOL(1)+(BLDG-182)+(BLDG-109NT))

C SLOCA-4 = SLOCA*A2*G*(-A)*(-C)
 ACC(4) = IE(3)*ABAR*CBAR*(BOOL(7)+PLOSP*A2)

C SLOCA-8 = SLOCA*G*A*(-C)
 ACC(5) = IE(3)*CBAR*(BOOL(4)+BOOL(5)+BOOL(6))

C SLOCA-9 = SLOCA*G*A*C
 ACC(6) = IE(3)*(BOOL(2)+BOOL(11)+(BLDG-182)+(BLDG-109NT))

C BLDG 182
 ACC(7) = IE(4)

C T4(LOSP) = LOSP*A2*(-A)*(-C)
 ACC(8) = IE(5)*ABAR*CBAR*A2

C T8(LOSP) = LOSP*A*(-C)
 ACC(9) = IE(5)*CBAR*BOOL(3)

C T9(LOSP) = LOSP*A*C
 ACC(10) = IE(5)*BOOL(1)

C T4(MISC) = TRANS*A2*G*(-A)*(-C)
 ACC(11) = IE(6)*BOOL(7)*ABAR*CBAR

C T8(MISC) = TRANS*G*A*(-C)
 ACC(12) = IE(6)*CBAR*(BOOL(4)+BOOL(10))

C T9(MISC) = TRANS*G*A*C
 ACC(13) = IE(6)*BOOL(2)

APPENDIX F

Core and Shield Response Plots

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-1 N-S Response Spectrum @ 1 SSE -- 40 sec t/h #3
Shield acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-2 N-S Response Spectrum @ 1 SSE -- 40 sec t/h #3
Core acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-3 N-S Response Spectrum @ 1 SSE -- 40 sec t/h #5
Shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:Accelerations are in ft/sec²

Figure F-4 N-S Response Spectrum @ 1 SSE -- 40 sec t/h #5
Core acceleration response spectra w 2% & 5% damping

Legend:

5% damping
2% damping

Notes:

Accelerations are in ft/sec²

Figure F-5 N-S Response Spectrum @ 2 SSE -- 40 sec t/h #3
Shield acceleration response spectra w 2% & 5% damping

Figure F-6 N-S Response Spectrum @ 2 SSE -- 40 sec L/h #3
 Core acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:Accelerations are in ft/sec²

Figure F-7 N-S Response Spectrum @ 2 SSE - 40 sec t/h #5
Shield acceleration response spectra w 2% & 5% damping

Figure F-8 N-S Response Spectrum @ 2 SSE -- 40 sec t/h #5
 Core acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-9 N-S Response Spectrum @ 3 SSE -- 40 sec t/h #3
Shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-10 N-S Response Spectrum @ 3 SSE -- 40 sec t/h #3
Core acceleration response spectra w 2% & 5% damping

Figure F-11 N-S Response Spectrum @ 3 SSE -- 40 sec t/h #5
Shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping
2% damping

Notes:

Accelerations are in ft/sec²

Figure F-12 N-S Response Spectrum @ 3 SSE -- 40 sec t/h #5
Core acceleration response spectra w 2% & 5% damping

Legend:
 5% damping
 2% damping

Notes:

Accelerations are in ft/sec²

Figure F-13 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft
 Primary shield acceleration response spectra w 2% & 5% damping

Legend:
 5% damping
 2% damping

Notes:

Accelerations are in ft/sec²

Figure F-14 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
 Primary shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping : —
 2% damping : - - -

Notes:

Accelerations are in ft/sec²

Figure F-15 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft
 Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-16 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-17 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft
(Core acceleration response spectra w 2% & 5% damping)

Legend:

5% damping

2% damping

Notes:Accelerations are in ft/sec²

Figure F-18 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Core acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
— 5% damping
--- 2% damping

Figure F-19 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft
Primary shield acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
— 5% damping
--- 2% damping

Figure F-20 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft
Primary shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping
2% damping

Notes:

Accelerations are in ft/sec²

Figure F-21 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping
2% damping

Notes:

Accelerations are in ft/sec²

Figure F-22 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-23 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft
Core acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-24 E-W Response Spectrum @ 1 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft
Core acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-25 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft
Primary shield elevation response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-26 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Primary shield acceleration response spectra w 2% & 5% damping

Figure F-27 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-28 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-29 E-W Response Spectrum @ 2 SSE --- 40 sec t/h #3 -- Elevation 5.4 ft
Core acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-30 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Core acceleration response spectra w 2% & 5% damping

Figure F-31 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft
Primary shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping
2% damping

Notes:

Accelerations are in ft/sec²

Figure F-32 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft
Primary shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-33 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-34 E-W Response Spectrum @ 2 SSE -- 40 sec. t/h #5 -- Elevation 35.3 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-35 E-W Response Spectrum @ 2 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft
Core acceleration response spectra w 2% & 5% damping

Figure F-36 E-W Response Spectrum @ 3 SSE -- 40 sec 1/h #3 -- Elevation 5.4 ft
Primary shield acceleration response spectra w 2% & 5% damping

Figure F-37 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #3 Elevation 35.3 ft
 Primary shield acceleration response spectra w 2% & 5% damping

Figure F-38 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping
2% damping

Notes:

Accelerations are in ft/sec²

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Figure F-39 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Inlet/Out shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-40 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #3 -- Elevation 5.4 ft
Core acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-41 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #3 -- Elevation 35.3 ft
Core acceleration response spectra w 2% & 5% damping

Legend:

5% damping ———

2% damping - - - - -

Notes:

Accelerations are in ft/sec²

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Figure F-42 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft
Primary shield acceleration response spectra w 2% & 5% damping

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Figure F-43 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft
Primary shield acceleration response spectra w 2% & 5% damping

Legend:

5% damping

2% damping

Notes:

Accelerations are in ft/sec²

Figure F-44 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #5 -- Elevation 5.4 ft Inlet/Outlet shield acceleration response spectra w/ 2% & 5% damping

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Figure F-45 E-W Response Spectrum @ 3 SSE --- 40 sec t/h #5 -- Elevation 35.3 ft
Inlet/Outlet shield acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% damping
2% damping

Figure F-46 E-W Response Spectrum @ 3 SSE --- 40 sec t/h #5 --- Elevation 5.4 ft
--- Acceleration response spectra w 2% & 5% damping

Notes:
Accelerations are in ft/sec²

Legend:
5% da 1g
2% da 1g

Figure F-47 E-W Response Spectrum @ 3 SSE -- 40 sec t/h #5 -- Elevation 35.3 ft
Core acceleration response spectra w 2% & 5% damping

END

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