Duke Power Company

MCGUIRE NUCLEAR STATION

IPEEE SUBMITTAL REPORT

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1. EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

In March 1982, Duke Power Company initiated a Probabilistic Risk Assessment (PRA) Study (Ref. 1.1) of the McGuire Nuclear Station, and this study was completed in July 1984. In 1988, Duke began a program to update this study to take into account a number of modifications to the plant and to take advantage of plant specific data and state-of-the-art methods. In 1991, Duke submitted this updated PRA (Ref. 1.2) to meet the requirements of Generic Letter 88-20 (Ref. 1.3) concerning the Individual Plant Examination (IPE) addressing internal events. The IPE Submittal Report (Ref. 1.4) explained that the McGuire PRA is a full-scope, level 3 PRA with complete analysis of external events in addition to internal events. External events have been included in the McGuire PRA studies beginning with the original study.

Consistent with the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities submittal plans outlined in the December 18, 1991 Duke letter (Ref. 1.5), and approved by the NRC letter of June 16, 1992 (Ref. 1.6), Duke Power Company provides herein the response to GL 88-20, Supplement 4 (Ref. 1.7). Included in this report (designated as the IPEEE Submittal Report) is a revision of certain sections of the McGuire PRA report. To facilitate the NRC staff review, the IPEEE information has been presented using the standard table of contents given in Table C.1 of NUREG-1407 (Ref. 1.8).

1.2 PLANT FAMILIARIZATION

The McGuire Nuclear Station is located in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina. It is on the banks of Lake Norman, impounded by Duke Power Company's Cowans Ford Dam Hydroelectric Station. The station consists of two four-loop Westinghouse pressurized reactors, each designed to generate 3411 MWt. The station was designed and constructed by Duke Power Company. The units were placed in commercial operation in December 1981 for Unit 1 and March 1984 for Unit 2.

The plant design incorporates the Standby Shutdown Facility (SSF), a totally independent means of achieving and maintaining safe shutdown conditions if the normal plant safety systems are unavailable.

The reactor containment is of the ice condenser type.

1.3 OVERALL METHODOLOGY

1.3.1 External Events Methodology

The evaluation of external events was performed in the original McGuire PRA report and its subsequent update with four events identified for a detailed review:

- Seismic Activity
- Fires
- Tornadoes
- Floods

In addition, NUREG-1407 requires a review of transportation and nearby facility accidents. (It should be noted that these events were also evaluated in the original PRA report, but their probabilities of occurrence were determined to be very low - <1E-08. Nevertheless, an evaluation using updated information is presented.)

A variety of methodologies were employed to derive the overall event frequencies for these events, as explained in detail in Sections 3.0, 4.0, and 5.0 of this report. In some cases, the findings from the original PRA studies did not require revision; otherwise, the information has been updated as necessary to support this examination.

1.3.2 Plant Model

The McGuire PRA is a full-scope analysis comprised of three parts and uses methods consistent with the PRA Procedures Guide (NUREG-2300) (Ref. 1.9). The Level I or "front end" analysis determines core damage sequences as a result of various internal and external events and places these sequences into plant-damage state bins. The Level II and III or "back end" analyses determine the effect of the accident sequences on containment and the resulting radiological releases to the public.

The basic models used for accident sequence development are event trees and fault trees. The event trees used in this analysis are functional event trees, with the top events defining the functions needed to protect the core. The end states of the functional event tree represent functional sequences. The event tree end states are also used to place accident sequences into plant-damage state bins. These bins are the transition from the front end analysis to the back end analysis.

The plant systems have been analyzed with detailed fault trees, generally to the component level. The level of detail in the model is defined by the level at which data is available.

This IPEEE study is primarily a Level I analysis which determines the event frequencies of external events. As with internal events, external events are input to the Level I plant model and their contribution to core-melt risk is determined. The Level II analysis involves the containment response to various accidents and core damage progression thereof and, thus, is not expressly influenced by external events. Rather, the external event impacts on active systems that affect containment performance (e.g., containment ventilation, spray, isolation, etc.) are addressed in this examination.

1.4 SUMMARY OF MAJOR FINDINGS

The major findings from this examination are that there are no unduly significant sequences (vulnerabilities) from external events. Tornadoes and seismic events are the most significant external event contributors to core-melt risk. For both hazards, the primary accident sequences involve a loss of off-site power with diesel-generator failures, thereby resulting in a loss of all ac power. There were no plant changes identified that would significantly reduce the risk from external events.

1.4.1 Core Damage Frequency Results

The results of the McGuire PRA report provide an estimate of plant severe accident risk and an understanding of the basis for this risk. The Core Damage Frequency (CDF) from external events as a result of the IPEEE evaluation is 3.0E-05 / yr., compared to 3.4E-05 / yr. estimated in the McGuire IPE report.

The contribution of the external events to the CDF and their comparison with the IPE values is shown in Table 1-1.

Tornado events make up 63% of the calculated external event CDF frequency. All of the tornado-initiated sequences are identical to non-recoverable loss of off-site power sequences. The tornado-induced loss of off-site power is followed by failures of the emergency power system. Emergency power system failures are dominated by failures of the diesel generators to run or start on demand. Diesel unavailability due to maintenance is also a significant contributor.

Seismic events comprise 36% of the calculated external event CDF frequency. Many of the dominant sequences involve a loss of off-site power followed by a failure of the emergency diesel generators. At low ground accelerations, diesel failures are due to random start, run, or maintenance failures. At ground levels above 0.5g, the diesel failures are predominantly seismic failures (diesel generator battery chargers, diesel oil tanks, dc control power, etc.). The loss of off-site power is assumed to be non-recoverable (a potentially conservative assumption). In addition, no credit is taken for recovering either diesel generator (another potentially conservative assumption).

The mean hazard curve generated by EPRI, specifically for the McGuire site, is used as the basis for this analysis. A sensitivity study was performed using the January 1989 Lawrence Livermore National Lab (LLNL) hazard curves for McGuire. The dominant accident sequences are comparable in their ranking with the EPRI curve results and do not add to or alter any of the insights of this analysis.

Internal fire events account for a relatively small contribution (2.3E-07 / yr.) to the overall external events core melt frequency.

1.4.2 Containment Performance Results

External event impact on containment performance has been examined from several perspectives, as follows:

- *Containment Structure* - The structure, penetrations, piping, and isolation valves all have median seismic acceleration capacities greater than 2.5g. The effects of airplane crashes and turbine-generated missiles were determined to be insignificant. No other hazard was identified that could challenge the containment structure.

- *Containment Isolation* - A screening analysis of containment penetrations was performed to determine which of these, if failed, could lead to significant release pathways. The seismic impact on containment isolation was evaluated by analyzing these penetrations and their associated piping, valves, supports, and isolation signals. They were found to be sufficiently rugged to withstand a seismic event. Effects of relay chatter were also considered. However, since none of McGuire's relays that would compromise safe shutdown functions qualify as "bad actors", this did not become a concern.

- *Containment Response* - External events were judged to have no significant impact on the containment performance model. An external event is modeled the same as an internal event with regard to containment response.

1.4.3 Vulnerability Findings

The basic finding of the evaluations summarized in this report is that there are no fundamental weaknesses or vulnerabilities with regard to severe accident risk at McGuire Nuclear Station.

TABLE 1 - 1

External Initiating Events Core Damage Frequency

	IPE Report (11/91)		IPEEE Report (6/94)	
	Core Damage Frequency (per year)	Percent of Total	Core Damage Frequency (per year)	Percent of Total
Initiating Event				
Seismic	1.4E-05	41.2%	1.1E-05	36.4%
Fires	8.1E-08	0.2%	2.3E-07	0.8%
Tornadoes	1.9E-05	55.9%	1.9E-05	62.9%
Ext. Flooding	<1.0E-08	<0.1%	<1.0E-08	<0.1%
Transportation & Nearby Facilities	--------	--------	--------	--------
Total External	3.4E-05		3.0E-05	

2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

The Individual Plant Examination Of External Events (IPEEE) for McGuire Nuclear Station was performed on the basis of the original McGuire PRA and its subsequent updates. This report summarizes the examination process for external events performed from 1982 - 1984 for the original McGuire PRA, the continuing process of updating the risk model which resulted in the updated PRA issued in 1991, and the results of the latest update to support the IPEEE.

The method of examination of external events used in the McGuire PRA and subsequent updates is the standard PRA method, with the enhancements described in Section 4 of the Generic Letter 88-20, Supplement 4. State-of-the-art probabilistic risk assessment (PRA) methods and current plant information were used in the original McGuire PRA and in the subsequent updates. The specific external events identified in GL 88-20, Supplement 4 have been addressed and are discussed in the pertinent sections. Comprehensive plant walkdowns have been performed to investigate and to incorporate the actual plant conditions in the examination. The basic event values involving random equipment failure, human error probabilities, and test and maintenance unavailabilities are compiled in the IPE analysis (Ref. 1.4).

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

Generic Letter 88-20, Supplement 4 identified four general purposes for each utility in performing the IPEEE. Duke Power Company has satisfied these as follows:

1. *Develop an Appreciation of Severe Accident Behavior* - Duke Power Company's initial staffing to enable large scale PRA and reliability studies in-house began in 1980. A severe accident analysis group was organized and charged with the responsibility to plan, conduct, and coordinate all proposed PRA studies and to maintain and update the plant PRA models as appropriate. In addition to PRA studies, this group is also utilized for engineering support involving severe accident input in such areas as emergency planning, plant design changes and plant operational problems.

   In conducting a full-scope PRA, personnel from the Severe Accident Analysis Group perform a majority of the PRA-related tasks. This core group is augmented by specialized expertise in mechanical, electrical and civil disciplines from other areas of the Nuclear Generation Department. In addition, the expertise of an operations engineer, assigned to support the PRA effort, is utilized to factor in operational insights on initiating events, accident sequence modeling, human reliability analysis and recovery actions. In the case of some specialized inputs, such as site seismology and equipment fragilities, outside expertise is utilized to complete the tasks.

The IPEEE effort was completed in-house, using the existing PRA work on external events augmented with the enhancements recommended in NUREG-1407.

2. *Understand the Most Likely Severe Accident Sequences* - The McGuire PRA report and the IPEEE evaluation have consistently shown the same dominant accident sequences from external events. Tornadoes and seismic events are identified to be the most significant external event contributors to core-melt risk. For both hazards, the dominant sequences involve a loss of off-site power with diesel-generator failures, thereby resulting in a loss of all ac power.

3. *Gain a More Qualitative Understanding of the Probabilities of Core Damage and Releases* - The plant systems have been analyzed with detailed fault trees, generally to the component level. The CAFTA computer code (Ref. 2.1) was used to solve the plant models and generate accident sequences in response to various internal and external events. The systems analysis also included the development of models needed to support the containment safeguards response. The accident sequence cut sets for core damage were coupled with the possible containment safeguards states, resulting in the final plant damage states. Thus, an understanding of the dominant core melt sequences and their consequences is understood on both a qualitative and quantitative basis.

4. *Reduce, if Necessary, the Overall Probabilities of Core Damage and Releases* - Whereas this examination of external events did not result in any major actions or modifications which could potentially reduce the overall core melt probability, several plant enhancements were identified during the development of the internal events portion of the IPE. These may be found in Section 3.0 of the McGuire IPE submittal report. The detailed IPEEE walkdown effort identified a few minor modifications to enhance the seismic adequacy of several components. These are listed in Table 3-3.

The Generic Letter Supplement also identified the issue of ensuring the technical adequacy of the IPEEE and validating its results. These are addressed as follows:

- The basis of the IPEEE report, the original PRA study and subsequent update, has received several stages of internal review. First, each of the major analytical tasks went through a peer review within the project team. Subsequently, it was reviewed by the project manager / engineering supervisor to ensure that the analyst had performed an adequate analysis and that it had gone through an appropriate peer review. Following the two levels of review performed within the project team, engineering personnel outside the PRA project team familiar with plant systems and accident sequences conducted a review of system models, underlying assumptions, system level results, and overall results. In parallel with the engineering review, the PRA draft report was reviewed by selected station personnel. The focus of this review was the reasonableness of underlying assumptions for system operation and operator actions. Besides the technical review of the PRA, management briefings were given to appraise key management personnel of the results and conclusions.

- The results of the IPEEE effort were given approximately the same level of review as the previous studies.

- Independent Review Teams (seismic and fire) were formed to perform a review of the IPEEE process and results. These teams consisted of senior level employees with experience in PRA methodology, seismic equipment qualification, fire protection, and systems engineering.

Thus, Duke Power has satisfied the objectives of the generic letter by its original PRA, subsequent updates, and the latest IPEEE effort. Duke Power has involved its staff to realize the maximum benefits from the program by involving its staff in all aspects of the examination.

2.3 GENERAL METHODOLOGY

The general methodology for examining external events is consistent with the methods presented in NUREG/CR-2300. The general approach used to develop the external event PRA is as follows:

- All natural and man-made external events were identified using other PRAs, NSAC/60 (Ref. 2.2), ANSI/ANS-2.12 (Ref. 2.3), and the aforementioned NUREG/CR-2300.

- The resulting events were screened in order to select significant events requiring further review. Twenty events were identified.

- A scoping analysis was performed on the remaining events. Four were identified that warranted a detailed quantification: earthquakes, floods, tornadoes, and fires.

This approach is presented in greater detail in the McGuire PRA report. (Note that this revised external events analysis also includes an updated review of transportation and nearby facilities accidents per NUREG-1407.)

The specific methodology for each hazard is discussed in Sections 3.0, 4.0, and 5.0.

2.4 INFORMATION ASSEMBLY

Many sources of plant documentation were used during the IPEEE process. These include the McGuire FSAR (Ref. 2.4), vendor seismic qualification design reports, vendor seismic qualification test reports, equipment specifications, plant drawings, vendor drawings, dynamic analyses of structures, in-structure response spectra, McGuire Probabilistic Seismic Hazard Evaluation (Ref. 2.5), structural design calculations, equipment anchorage design calculations, flow diagrams, computer codes, air traffic information, evacuation plans, and operating procedures. Additional sources of information related to the fire review are listed in Section 4.11.

The original PRA report included the then-current plant design documents, operating procedures, Tech. Specs., and plant configuration. The subsequent revision to the PRA used updated information as appropriate. The impact of external events to the overall CDF was also considered.

Coordination activities of the IPEEE teams among the external events are handled by Duke Power's Severe Accident Analysis group which is responsible for the McGuire PRA. Individuals from this group were on all the teams and were responsible for coordination and the final results. As an example, any potential for seismically induced fires were communicated between the fire and seismic teams.

3. SEISMIC ANALYSIS

The seismic PRA methodology was utilized to perform the seismic IPEEE. Because the current McGuire PRA report includes a seismic analysis, the procedure for completing the seismic portion of the IPEEE was followed using the recommendations given in NUREG-1407, Section 3.1.2.

3.1 SEISMIC PRA

This section describes the methods used to estimate the contribution to public-health risk from earthquakes at McGuire. The analysis uses a methodology consistent with NUREG/CR-2300.

The first step in the analysis was to obtain a site-specific seismic hazard curve. This curve represents the likelihood that ground motions of varying magnitudes would occur. Fragility curves were then developed for key components and structures at McGuire Unit 1. These fragility curves were used to determine the conditional probability of failure as a function of ground acceleration. An event tree was then developed, along with supporting top logic and system fault trees. The event tree was used to develop a Boolean expression for the event sequences of interest. The final step involved combining the fragility curves, using the Boolean expression, and convoluting this failure probability with the site seismicity.

The seismic model used in this evaluation has been updated to reflect changes to the plant, fragility information, and fault tree logic.

3.1.1 Hazard Analysis

Figure 3-1 presents the results of the McGuire seismic hazard study. The results are in the form of hazard curves for peak acceleration. The study was performed using the Seismicity Owners Group (SOG) methodology, developed by the Electric Power Research Institute (EPRI), for seismic hazard analysis of nuclear power plant sites in the Central and Eastern United States (CEUS) (Ref. 3.1). The tectonic interpretations were prepared by six earth science teams, also under the sponsorship of SOG. This methodology (EQHAZARD) has been reviewed by the U.S. Nuclear Regulatory Commission and was found to be an acceptable method for calculating seismic hazard at nuclear power plants in the CEUS.

In the SOG methodology, models of future earthquake occurrences are deduced from tectonic theories, geologic evidence, analogies with other regions, and historical seismicity. These models consist of seismic sources and seismicity parameters for each source:

- A seismic source is a geometric representation of either a tectonic feature (or group of features) that is capable of producing earthquakes or an area of seismic activity with no clear association with any tectonic feature.

- The seismicity parameters 'a' and 'b' of a seismic source describe the recurrence rates of different magnitude earthquakes in the source where 'a' is a measure of seismic activity per unit area, and 'b' is a measure of the relative frequency of large versus small events. Magnitude is characterized by the body-wave magnitude m_b. A third seismicity parameter, m_{max}, indicates the largest magnitude that may occur in the seismic source. Earthquakes with magnitudes lower than magnitude m_{min} are neglected because they do not cause damage to engineered structures.

In addition to the above model of future seismicity, it is necessary to estimate the ground-motion amplitude at the site, given the magnitude of the earthquake and its distance to the site. This estimation is performed using an attenuation function or ground-motion model.

Seismic sources, their seismicity parameters, and the attenuation function are combined using the total-probability theorem. The result is a seismic hazard curve that gives annual exceedance probability (hazard) as a function of ground-motion amplitude.

Uncertainty in seismic hazard arises because there are alternative theories on the causes and characteristics of earthquakes in the Central and Eastern United States (scientific uncertainty), and because there is a limited amount of data (statistical uncertainty). These uncertainties translate into uncertainties in the input parameters (i.e., which geologic features are seismic sources, what are their parameters, what is the attenuation function). The SOG methodology provides a framework for documenting the sources of uncertainty in the input parameters, propagating uncertainties through the hazard calculations (thus obtaining the seismic hazard and its total uncertainty), and evaluating the contribution of each type of parameter uncertainty to the total uncertainty.

To determine uncertainty of the causes and characteristics of earthquakes in the Central and Eastern United States, six earth-science teams were asked to identify all potential seismic sources in that area, and evaluate their seismic potential and seismicity parameters. Furthermore, each team was asked to consider alternative hypotheses and state the degree of uncertainty on the seismic potential and the seismicity parameters of the sources identified. These interpretations serve as input to the EQHAZARD computer programs.

3.1.2 Review of Plant Information and Walkdown

This section discusses the seismic design of the plant, sources of information used in the fragility analysis, and the confirmatory walkdowns that were done in support of the PRA and this IPEEE report.

3.1.2.1. Plant Information

The McGuire systems and components which are essential to the prevention or mitigation of accidents and could affect the public health and safety were designed to enable the facility to withstand the effects of natural forces including earthquakes. The plant was designed to withstand both an Operating Basis Earthquake (OBE) and a Safe Shutdown Earthquake (SSE). The structural design criteria for the SSE was based on 0.15g and the OBE on 0.08g peak horizontal ground accelerations for all Seismic Category I structures. Vertical accelerations equal to two-thirds of the corresponding horizontal values were used for both the OBE and SSE.

Almost all the Seismic Category I structures are founded on competent rock or concrete fill extending to rock. The refueling water storage tanks are founded on residual soils. These soils are produced from weathering of crystalline rocks. The soils profile shows soils grading into stiff to very stiff micaceous silts, grading into partially weathered rock. The diesel generator fuel tanks are founded on compacted fill extending to rock or weathered rock. The residual soils and compacted fill are considered stable against the seismic events of interest. The effects of soil-structure interaction are considered negligible for the Category I structures. The ground response spectra used in the design of the structures were developed by N. M. Newmark. Artificial earthquake time-history records whose response spectra essentially envelop the smoothed ground response spectra were created for use in developing the in-structure floor response spectra used for equipment qualification.

The plant structures and equipment were originally divided into two categories according to their function and the degree of integrity required to protect the public. These categories are Category I and non-Category I. McGuire Nuclear Station structures, systems, and components important to safety, as well as their foundations and supports, were designed to withstand the effects of an OBE and an SSE and were designated as Seismic Category I. In addition, the foundations and supports for all Category I structures and supports are designed for Category I criteria. The major seismic Category I structures include the following:

- Reactor Building
- Auxiliary Building
- Diesel Generator Building
- Standby Nuclear Service Water Intake, Discharge, and Overflow Spillway

Structures, equipment and components which are important to plant operation, but are not essential for preventing an accident which would endanger the public health and safety, and are not essential for the mitigation of the consequences of these accidents, are classified as non-Category I. The Turbine Buildings are non-Category I structures. The Turbine Buildings are designed against collapse onto Category I structures due to SSE loads. Examples of non-Category I equipment include the offsite power, the station power transformers, the condenser hotwell and the hydrogen igniters. Non-safety related equipment routed or located in seismic Category I buildings that present interaction concerns with safety equipment is designed to withstand the effects of the SSE so that the function of important equipment is not compromised.

3.1.2.2. Information Sources

A structural analysis consultant, Structural Mechanics Associates, was used to develop the structural and equipment fragilities for the original fragility analysis. For the most part, results of existing analyses and evaluations of structures and equipment for the McGuire plant were utilized in this study. As part of the evaluation, some limited analysis based on original design analysis loads was conducted to determine the expected seismic capacities of the important structures. This section summarizes the important sources for the fragility analysis.

There was a general lack of detailed information available on the seismic capacity of specific McGuire structures and equipment beyond their design basis. This occurs because existing codes and standards do not require determination of ultimate seismic capacities, either for structures or equipment qualified by analysis or for equipment or components qualified by testing. Therefore, most median safety factors, estimates of variability, and conditional frequencies of failure estimated in this study are based on existing analysis and qualified engineering judgment and assumptions.

Structural Response and Capacity

The results from the existing design basis dynamic analyses of the important structures were extensively used in this study. These were supplemented as required, for example, to provide estimates of load redistributions resulting from localized failures caused by loading beyond the SSE.

Levels of conservatism associated with the method of analysis used in design were estimated such that safety factors reflecting this analytical conservatism could be estimated for the building structures and for the seismic excitation of equipment mounted within the building.

In general, detailed structural design calculations were not reviewed for the fragility analysis, but rather the design criteria as defined in the FSAR were reviewed. Some ultimate load capacity analyses were conducted which served as a basis for estimating the median factor of safety on structural resistance to the SSE.

Seismic Category I Piping and Equipment Response and Capacity

For most of the safety-related piping and equipment, information on analysis methods was available in summary form in the FSAR. Seismic response information was obtained from the seismic qualification reports for specific components. In some cases, such as for piping, only the seismic analysis requirements and stress acceptance criteria were known. Safety factors for response and structural or functional capacity were estimated from existing information. No new analyses were conducted.

In-structure response spectra for all Category I structures were generated during the design process. From these typical floor response spectra and knowledge or estimates of equipment fundamental frequencies, an estimate is made of the peak equipment response. The peak equipment response estimate is then compared to the dynamic response or equivalent static coefficient used in design to determine a median safety factor on response.

Capacity factors are derived from several sources of information: plant-specific design reports, test reports, generic fragility test data from military test programs, and generic analytical derivations of capacity based on governing codes and standards. The structural and functional failure modes are considered in developing capacity factors for piping and equipment. Equipment and piping design reports delineate stress levels for the specified seismic loading plus normal operating conditions. Where the equipment fails in a structural mode (i.e., pressure boundary rupture or loss of support), the median capacity factor and its variability are derived in the same manner as for structures considering strength and energy absorption (ductility). In cases where equipment must function, the capacity factor is derived by comparing the equipment functional failure (or fragility) level to the design level of seismic loading. There are some fragility test data on generic classes of equipment that have been utilized in hardened military installations. The equipment was off-the-shelf without special shock resistant design and is similar to nuclear power plant equipment. These data provide estimates of the fragility levels, and thus, safety factors can be developed for the specified design earthquake. Fragility levels are not normally determinable from equipment qualification reports, but the achieved test levels can be utilized to update generic fragilities derived from the test data.

For developing equipment fragilities, a number of plant specific and generic information sources were used. Some of these information sources are termed "plant specific" since they pertain to specific equipment within the McGuire plant. The other information sources are termed "generic" since they constitute data generated for similar types of equipment or are definitions of design requirements, in lieu of actual design results. Plant specific sources are preferred since they have been generated for the specific items in question and their uncertainty level is reduced from those of the generic sources. Several sources of information are highlighted below.

Seismic Qualification Analysis Reports

The majority of the fragility levels for critical McGuire equipment were developed from the review of seismic qualification analysis reports. Westinghouse provided qualification report summaries for most of the NSSS equipment items, and fragility levels were calculated based on these summaries. In some cases, the Westinghouse supplied data were based on a generic analysis where generic spectra, which enveloped the response spectra for several plant sites, had been used for the loading. In these cases, the stresses have been scaled down to reflect the response for the McGuire site, and thus, these cases essentially constitute a plant specific analysis.

Seismic Qualification Test Reports

Some examples of test reports for equipment qualified by testing were reviewed. Qualification test reports, by themselves, cannot be utilized to develop fragility relationships unless the equipment has been tested to increased vibration levels up to failure. Consequently, most equipment qualified by test was treated generically with the test qualification report data (when reviewed) being considered as part of the population of test data on similar generic equipment.

Past Earthquake Experience

Past earthquake experience is valuable for establishing fragilities for equipment which have historically been vulnerable. Most equipment survives without any apparent damage, and the historic experience must be treated the same as a qualification test. Earthquake experience has typically been used to estimate fragility levels for off-site power systems and non-seismically qualified equipment.

Specification on the Design of Equipment

Specifications for seismic qualification of McGuire equipment were provided by Duke Power. In cases where plant specific qualification reports were not reviewed, knowledge of the vendor requirements plus generic fragility and qualification test data were combined to develop fragility descriptions.

3.1.2.3 Walkdowns

Plant walkdowns are considered to be an important part of the seismic risk assessment. In support of this assessment, a number of walkdowns were conducted. Walkdowns were performed to support the development of the initial McGuire PRA which included external events. The initial study was completed in 1984. Walkdowns were also conducted to support the update of the McGuire

PRA, submitted in 1991. As a part of the IPEEE effort, extensive walkdowns were conducted throughout 1993 consistent with the guidelines described in EPRI NP-6041 (Ref. 3.2). Walkdowns were conducted on both units of the plant. Approximately 200 mechanical and electrical components and 270 valves were walked down on McGuire Unit 1. Approximately 160 mechanical and electrical components and 220 valves were walked down on McGuire Unit 2. Moreover, general area reviews were also conducted within the plant to evaluate bulk distribution systems. Walkdowns were conducted inside containment for each unit, focusing on equipment list items inside containment as well as "containment performance" issues. Much more extensive walkdowns were performed outside of containment. Areas surveyed include the Auxiliary Building, Diesel Generator Building, and the Main Steam & Feedwater Isolation Compartments (Doghouses). The purposes of these walkdowns were to confirm the validity of the earlier equipment fragility assessments, to verify seismic adequacy of equipment anchorage, and to identify any other seismic concerns such as potential seismic spatial interactions in the "as-built" plant configuration.

The walkdowns also included a review of the potential for fires and flooding in the plant resulting from a seismic event. In addition to the Seismic Review Team's (SRT) surveillance, any potential fire sources resulting from a seismic event were also communicated to the SRT by the independent review of the Fire Review Team which was also focusing on the issue. To address the flooding issue, potential for ruptured vessels or piping that could spray, flood or cascade onto essential equipment in vulnerable areas of the plant was considered during the walkdowns.

These latest walkdowns were conducted by civil/structural engineers, trained in the "Seismic Margin Methodology" outlined in EPRI NP-6041, PRA team members, station system and equipment engineers, and supporting station craft personnel. The walkdown was conducted using the procedures and documentation forms recommended in EPRI NP-6041.

In preparation for the walkdowns, a significant amount of time was spent developing equipment lists and new in-structure response spectra, familiarizing the walkdown team with the plant design information, marking up plant general arrangement drawings with the location of the components to be evaluated, reviewing for low ruggedness relays, and performing equipment anchorage evaluations. Following the walkdowns, resolution of outliers which could not be screened out using the walkdown screening criteria and remaining anchorage evaluations were performed.

The Seismic Margin Methodology guidelines provide generic conservative estimates of ground motion below which it is generally not necessary to perform a seismic margin review for particular elements. Therefore, for a given ground motion level, these guidelines list the elements which should, in general, be "screened out" from margin review because of their generically good performance in earthquakes or seismic simulation tests at or above this level. This "screening out" is contingent on verifying during the walkdown that the equipment meets the caveats provided to insure that it is representative of equipment included in the earthquake experience data. In addition to the screening guidelines for the equipment, an anchorage assessment must also be conducted to verify that the anchorage is adequate for the specified ground motion. The detailed documentation of the screening evaluations, anchorage evaluations, and other supporting information has been compiled and filed as McGuire calculation MCC-1535.00-00-0004.

A NUREG/CR-0098 median ground response spectrum (Ref. 3.3) anchored at 0.3g was used for the review level earthquake for the McGuire site as recommended in NUREG-1407. New in-structure response spectra were developed for the Auxiliary Building using the review level earthquake. This work is filed as McGuire calculation MCC-1139.10-00-0248. For the Reactor Building, the seismic demand was estimated by scaling the Safe Shutdown Earthquake (SSE) demand estimates upward for the increase of the review level ground response spectrum over the SSE ground response spectrum.

The walkdown served to confirm the structure and equipment fragilities contained in the McGuire PRA. Some minor adjustments were made to the equipment fragilities as a result of this latest review. The fragility values for the Containment Spray and Residual Heat Removal Heat Exchangers were lowered based on the walkdown and subsequent anchorage evaluation. The fragility of the Residual Heat Removal Pump was raised due to an error in the previous fragility assessment. Four new components were also added to the list requiring fragility evaluations. These components are the Main Control Boards, Auxiliary Shutdown Panels, Motor Driven Auxiliary Feedwater Pump Control Panels, and Turbine Driven Auxiliary Feedwater Pump Control Panels. Several components which screened out using the seismic margin criteria have slightly lower HCLPF's based on previous fragility evaluations for these components. The margin criteria review indicates that the previous fragility evaluations for these components may be conservative but these fragilities were not revised to remove the conservatism. None of these fragility changes were significant to the risk results. Table 3-1 lists the structure and equipment fragilities. Since the equipment list was developed using both PRA and EPRI NP-6041 equipment selection criteria, more equipment was actually walked down and evaluated than necessary to validate the PRA.

In addition to the seismic fragility validation, a few minor seismic concerns such as spatial interactions were identified. These are listed in Table 3-3. None of these items were significant to the risk results.

The scope of the relay chatter review for McGuire is consistent with the site's seismic margin review level earthquake classification as defined in Table 3.1 of NUREG-1407. McGuire is in the "focused scope" bin. The focused scope evaluation is limited to a review of low seismic ruggedness relays or "bad actors," as found in EPRI NP-7148-SL, Appendix E (Ref. 3.4). A limited number of "bad actors" were found associated with the IPEEE equipment list but they were serving an alarm function rather than a control function. Therefore, no further evaluation was necessary.

All major Category I structures are founded on continuous rock or concrete fill extended to continuous rock. The Refueling Water Storage Tanks and Diesel Generator Fuel Oil Storage Tanks are founded on residual soils or compacted backfill. The Standby Nuclear Service Water Pond Dam also consists of compacted backfill. Soil tests performed on these soils during construction indicate that these soils are not susceptible to liquefaction.

3.1.3 Analysis of Plant System and Structure Response

This section describes the methods used for the fragility analysis. Fragility analysis consists of determining the ultimate capacity of structures and equipment to withstand a peak ground acceleration. The approach used in assigning peak ground acceleration capacities to safety-related structures, equipment, and other components was to first determine the median factor of safety against failure and its statistical variability under the safe shutdown earthquake (SSE). From this safety factor and variability, the median ground acceleration capacity and its variability were determined. For non safety systems, capacities were calculated and then keyed back to the SSE for results presentation.

The factor of safety of a structure or component is defined as the resistance capacity divided by the response associated the SSE of 0.15g effective peak acceleration. The development of seismic safety factors associated with the SSE is based on consideration of several variables. The variability of dynamic response to the specified acceleration and the strength capacity of the structure or equipment component are the two basic considerations in determining the variability in the factor of safety. Several variables are involved in determining both the structural response and the structural capacity, and each such variable, in turn, has a median factor of safety and variability associated with it. The overall factor of safety is the product of the factors of safety for each variable. The median of the overall factor of safety is the product of the median safety factors of all the variables. The variabilities of the individual variables also combine to determine that of the overall safety factor.

Variables influencing the factor of safety on structural capacity to withstand seismic induced vibration include the strength of the structure compared to the design stress level, the inelastic energy absorption capacity (ductility) of a structure or its ability to carry load beyond yield, and the earthquake duration to account for the expected duration compared to that assumed in determining the energy absorption factor. The variability in computed structural response for a given effective peak free-field ground acceleration is made up of many factors. The more significant factors include variability in (1) ground motion and the associated ground response spectra for a given peak free-field ground acceleration, (2) energy dissipation (damping), (3) structural modeling, (4) method of analysis, (5) combination of modes, (6) combination of earthquake components, and (7) soil-structure interaction.

The overall safety factor for equipment and other plant components is derived from similar factors for the component. However, their response also depends on the building in which they are located and their location within the building. Therefore, the overall safety factor for components is made up of component strength capacity relative to the floor acceleration, earthquake duration, component response, and building response that resulted in the floor spectra used in the component design.

The ratio between the median value of each of these factors and the value used in design of the McGuire plant and the variability of each factor are quantitatively estimated for various structures and components using available test data for McGuire, limited analysis, engineering judgment and experience in the analysis of nuclear power plants and components.

The derivation of each factor considered variability. When combining these median factors from each parameter, variabilities were also combined to determine the variability in overall safety factor. From this overall safety factor, the median acceleration capacity, or peak ground acceleration at failure, was determined by multiplying the safety factor by 0.15g, the SSE ground motion.

$$\hat{A} = F_E * A_{SSE}$$

where:

$\hat{A}$ = median acceleration capacity

A_{SSE} = peak ground acceleration of the SSE

$$F_E = F_{EC} * F_{ER} * F_{ED} * F_{SR}$$

where:

F_{EC} = capacity factor of safety for the equipment relative to the floor acceleration used for the design

F_{ER} = factor of safety inherent in the computation of equipment response

F_{ED} = earthquake duration factor of safety associated with the predicted number of strong motion cycles within a seismic event

F_{SR} = factor of safety in the structural response analysis that resulted in floor spectra for equipment design

Definition of Failure

For purposes of this study, seismic Category I structures are considered to have failed when inelastic deformations of the structure under seismic load potentially interfere with the operability of equipment attached to the structure. These limits on inelastic energy absorption capacity (ductility limits) are estimated to correspond to the onset of significant structural damage, not necessarily structure collapse. Piping, as well as electrical, mechanical, and electromechanical equipment vital to mitigating the effect of earthquakes are considered to fail when they can no longer perform their designated functions. Relay chatter is an example of a functional failure for an electrical component. Also, rupture of pressure boundaries are considered failures. For active equipment, the functional failure definition will usually govern as equipment pressure boundaries are usually very conservatively designed for equipment such as pumps and valves.

Seismic-induced fragility data are generally unavailable for specific plant components. Therefore, fragility curves must be developed primarily from analysis combined heavily with engineering judgment supported by very limited test data. Such fragility curves will contain a great deal of uncertainty, and it is imperative that this uncertainty be recognized in all subsequent analyses. Because of this uncertainty, great precision in attempting to define the shape of these curves is unwarranted. Therefore, a procedure which requires a minimum of information, incorporates uncertainty into the fragility curves, and easily enables the use of engineering judgment, was used in this study.

The entire fragility curve for any mode of failure and its uncertainty can be expressed in terms of the best estimate of the median ground acceleration capacity, $\hat{A}$, times the product of random variables. Therefore, the ground acceleration, A, corresponding to failure is given by:

$$A = \hat{A}\, \varepsilon_r \varepsilon_u$$

in which ε_r and ε_u are random variables with unit median representing inherent randomness (failure fraction) about the median and the uncertainty (probability) in the median value, respectively. Both ε_r and ε_u are assumed to be lognormally distributed with logarithmic standard deviations of β_r and β_u, respectively. The lognormal distribution can be justified as a reasonable distribution since the statistical variation of many material properties and the seismic response variables may reasonably be represented by this distribution. In addition, the central limit theorem states that a distribution consisting of products and quotients of distributions of several variables tends to be lognormal even if the individual distributions are not lognormal.

3.1.4 Evaluation of Component Fragilities and Failure Modes

The seismic capacities of plant structures and components were developed by Structural Mechanics Associates. The study (Ref. 3.5) gives a detailed description of how the seismic capacities were derived. The seismic capacity of the nuclear service water retaining dam was developed by Dr. Daniel Veneziano of MIT, a consultant to Law Engineering Testing Company. The results of that study are reported in References 3.6 and 3.7. The seismic capacities are presented in the final form of fragility curves, which express the conditional probability of failure as a function of ground acceleration. As mentioned in Section 3.1.2, several of the original fragilities were updated as a result of the IPEEE review. Structures with a median fragility of 2.5g or greater and components with a median fragility of 2.0g or greater are screened out of the model. Table 3-1 lists the fragilities of the remaining structures and components in the model. Values for the "High Confidence of Low Probability of Failure", or HCLPF, were not provided but can be determined with the following equation:

$$HCLPF = \hat{A}\ \exp[-1.65\,(\beta_r + \beta_u)]$$

In keeping with our previous definition, failure of structures is defined as any deformation sufficient to interfere with the operation of the attached equipment. For other equipment, failure is defined as pressure boundary rupture or loss of function.

The seismic analysis uses the best estimate of the median ground acceleration capacity, $\hat{A}$, along with two measures of uncertainty, to represent a fragility curve. As mentioned above, one measure of uncertainty is β_r, the logarithmic standard deviation associated with the underlying randomness of the component or structure. The other measure of uncertainty is β_u, the logarithmic standard deviation associated with the uncertainty of the median capacity.

3.1.5 Analysis of Plant Systems and Sequences

This section describes the process used to calculate the seismically-induced core-melt frequency.

3.1.5.1 Seismic Event Tree

The first step in determining the seismically-induced core-melt frequency is the creation of an event tree. The event tree for the McGuire seismic analysis is shown in Figure 3-2. The tree is structured similarly to the internal initiator event trees and contains the same functional top events. A small-break LOCA, event Qs, will occur if support systems to the reactor coolant pump seals are lost. Event U for the small-break LOCA varies for different branches of the event tree. If secondary side heat removal (SSHR) is available (indicated by a success of event B), injection is needed from one of the four high head pumps. If SSHR is not available, in addition to requiring the NV and NI pumps, a requirement for feed-and-bleed cooling path also exists. All successful U event sequences address event X, recirculation cooling.

If no LOCA exists following the seismic event, the core must still be cooled. If secondary side heat removal fails, decay heat removal must be achieved by performing a primary feed-and-bleed.

The components of the event tree are explained further:

Event B: Secondary Side Heat Removal Maintained

Failure of secondary side heat removal occurs due to a loss of the auxiliary feedwater pumps or a loss of the water sources. Auxiliary feedwater is also failed if the Auxiliary Building structure or its blockwalls fail.

Event Qr: Pressurizer Relief Valves Close After Opening

Event Qr is modeled as one event, TNCOSRVDEX. This event is a pressurizer safety relief valve failing to reseat after relieving liquid.

Event Qs: NC Pump Seal Integrity Maintained

Loss of support systems to the reactor coolant (NC) pump seals can lead to an NC pump seal LOCA. This LOCA occurs if both component cooling and seal injection fail. Seismic failure of the Auxiliary Building structure or the respective control panels, as well as a loss of all power or a loss of RN, would result in the failure of these support systems.

Event P: Bleed Path Established

Event P is successful operation of the PORVs during feed-and-bleed cooling. Failure of the PORVs occurs on a failure of instrument air, control power, the Auxiliary Building structure, or the operators failing to establish feed-and-bleed cooling. A loss of off-site power, which occurs at relatively low ground accelerations, causes a loss of normal instrument air to the PORVs. Failure of the backup nitrogen supply to the PORVs results in a total loss of feed-and-bleed cooling.

Event U: High Pressure Injection Established

Success criterion for event U is flow from any one of the NI or NV pumps. Failure of event U occurs due to a loss of power, failure of the pumps, failure of the respective control panels, failure of the FWST, or a loss of nuclear service water.

Event X: Long-Term Core Heat Removal Established

Failure of event X occurs due to failure of the ND pumps or heat exchangers, loss of component cooling, failure of the respective control panels, loss of power, or operators failing to establish recirculation resulting in a loss of recirculation.

3.1.5.2 Event Tree Sequences

Depending upon the success or failure of the above events, the accident sequences leading to a core melt are determined. These are described below:

Sequence CQsX

Sequence CQsX involves a seismically-induced failure of NC pump seal cooling and a failure of long-term heat removal. Because SSHR succeeds, a late core melt results. The seal failure produces leakage equivalent to a small-break LOCA.

Sequence CQsU

Sequence CQsU involves a seismically-induced failure of NC pump seal cooling and a failure of NI and NV in the injection phase. The seal failure produces the equivalent of a small-break LOCA. Because SSHR succeeds, the core melt is late.

Sequence CBX

Sequence CBX is a seismically-induced failure of SSHR and a failure of ND in the recirculation phase.

Sequence CBU

Sequence CBU is a seismically-induced failure of SSHR and failure of NI and NV in the injection phase.

Sequence CBP

Sequence CBP is a seismically-induced failure of SSHR and the failure to establish a bleed path via the PORVs.

Sequence CBQsX

Sequence CBQsX involves a seismically-induced failure of SSHR, failure of NC pump seal cooling, and failure of ND in the recirculation phase. The seal failure produces leakage equivalent to that of a small-break LOCA.

Sequence CBQsU

Sequence CBQsU involves a seismically-induced failure of SSHR, failure of NC pump seal cooling, and failure of NI and NV in the injection phase. The seal failure produces leakage equivalent to that of a small-break LOCA.

Sequence CBQsP

Sequence CBQsP is a seismically-induced failure of SSHR, failure of NC pump seal cooling, and the failure to establish a bleed path via the PORVs. The seal failure produces leakage equivalent to that of a small-break LOCA.

Sequence CBQrX

Sequence CBQrX is a seismically-induced failure of SSHR, a stuck open safety relief valve, and a failure of ND in the recirculation phase.

Sequence CBQrU

Sequence CBQrU is a seismically-induced failure of SSHR, a stuck open safety relief valve, and a failure of NI and NV in the injection phase.

3.1.5.3 Seismic Fault Tree Solution

After developing the event tree above, a fault tree was created to determine the various possible accident scenarios (cut sets) for the event tree sequences. The complete seismic fault tree is shown in Appendix A. A listing of the tree's basic events are given in Table 3-2. The fault tree was solved used the CAFTA computer code. The resulting cut sets were reviewed and edited to remove invalid and non-minimal cut sets from the solution. To account for containment performance considerations, failures and non-failures of the Containment Air Fans and Containment Spray System components were added to the remaining cut sets. These were then loaded into the SEISM computer code (Ref. 3.8) to determine the overall probability for a seismically induced core melt.

The SEISM methodology is similar to the "Zion method," which is described in the PRA Procedures Guide and used in the Zion Probabilistic Safety Study (Ref. 3.9). The only difference is in the manner of propagating the uncertainties in the component fragilities through the logic models. SEISM uses Monte Carlo simulation, whereas the Zion method uses discrete-probability-distribution techniques, in propagating the uncertainties. Monte Carlo simulation is shown in NUREG/CR-3263 (Ref. 3.10) to outperform discrete probability-distribution techniques.

The plant seismicity curve, component fragilities, and event sequences are inputs to SEISM in the calculation of the frequency of a seismically-induced core-melt. Component fragilities are combined, using the event sequences, to obtain cut set fragilities. A third-order approximation of the sum of the cut set fragilities determines the plant fragility. The plant fragility is convoluted with the plant seismicity curve to derive the frequency of a seismically-induced core-melt.

SEISM uses this process in two stages. In the first stage, best estimate values are used for the component fragilities and plant seismicity. This stage calculates a best estimate of the core-melt frequency. In the second stage, the component fragilities are obtained by randomly sampling fragility values from the family of curves of each component. These fragility values are used as the component fragilities in calculating a sampled core-melt frequency. This stage is repeated many times to obtain a sampled core-melt frequency distribution.

The SEISM output is shown in Table 3-4. This table gives the percent contribution of each cut set to the total seismically-induced core-melt frequency and shows how the frequency of each cut set is distributed among the acceleration intervals. (Components of the cut sets prefixed with a minus (-) sign indicate a *non*-failure of that component.) The primary cut sets are dominated by a loss of off-site power followed by a failure of the emergency diesel generators. At low ground accelerations, diesel failures are due to random start, run, or maintenance failures. At ground levels above 0.5g, the diesel failures are predominantly seismic failures (diesel generator battery chargers, diesel oil tanks, dc control power, etc.). The loss of off-site power is assumed to be non-recoverable (a potentially conservative assumption). In addition, no credit is taken for recovering either diesel generator (another potentially conservative assumption).

The probability for a seismically-induced core-melt was calculated by SEISM to be 1.1E-05 / yr. This value represents 36% of the external event contribution to the overall plant core-melt probability.

Note that the McGuire seismic hazard curve was developed up to an acceleration level of 1000 cm/sec^2 (1.02g). Because the probability of exceedence is approximately 3E-07 at this point, it is determined that extending the curve beyond 1.02g would not significantly affect the overall core-melt frequency results.

3.1.6 Analysis of Containment Performance

The external event impact on containment performance has been examined from several perspectives:

- Containment Structure

  Structural Mechanics Associates (Ref. 3.5) performed an analysis of the Reactor Building, steel containment vessel, and containment internal structures. The results of this review were as follows:

Structure	Failure Mode	Median Fragility
Reactor Building	Shear Failure of Wall	2.8g
Steel Containment Vessel	Buckling of Sidewall	9.0g
Internal Structures	Flexural Failure of Crane Wall	3.1g

  Based upon the above results, seismic failure of the containment structure is not considered a credible event.

- Containment Isolation

  For the McGuire PRA a screening analysis of containment penetrations was performed to determine which of these, if failed, could lead to significant release pathways. The details of this analysis may be found in PRA Appendix A.9. The seismic impact on containment isolation was evaluated by analyzing these penetrations and their associated piping, valves, supports, and isolation signals. Piping, valves, and supports were determined by Structural Mechanics Associates to have median fragilities greater than 2.5g. The upper and lower containment hatch inflatable door seals and latches were also reviewed.

  The containment isolation signals are generated from the Solid State Protection System (SSPS) to the Emergency Safeguards Features Actuation System (ESFAS). The cabinets housing this equipment were evaluated for functional ruggedness, resulting in a median fragility of 1.54g. Likewise, the respective panelboards and motor control centers providing power to actuate the valve solenoids and motors were analyzed. Their median fragilities were 1.66g and 1.68g, respectively. The above equipment was also evaluated via confirmatory walkdowns (see Section 3.1.2.3).

Effects of relay chatter were also considered. However, since none of McGuire's relays that would compromise safe shutdown functions qualify as "bad actors", this did not become a concern.

- Containment Response to Accident,

  As mentioned in Section 3.1.5.3, failures and non-failures of the Containment Air Return Fans and Containment Spray System components were added to the resulting seismic fault tree cut sets to account for containment performance considerations. The hydrogen mitigation (glow plug igniter) systems and ice baskets & doors were reviewed during the containment walkdown process.

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

USI A-45 "Shutdown Decay Heat Removal Analysis"

The decay heat removal capability has been addressed in Section 6.0 of the IPE report. The calculated annual core-melt frequency due to failure of decay heat removal systems for external initiators is approximately 1.0E-05. Therefore, the McGuire decay heat removal systems exhibit high resistance for external events. Therefore, this issue should be considered resolved for McGuire.

GI-131 "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants"

This issue was addressed by analysis of the system in 1985. This documentation is filed as McGuire calculation MCC-1117.04-10-0001. The seismic analysis indicated the restraints on the McGuire system are adequate to prevent seismic interaction and breach of the pressure boundary. Therefore, GI-131 is considered closed.

Other Seismic Safety Issues

- Eastern U. S. Seismicity Issue

  Significant uncertainty exists concerning seismic hazard curves. The mean hazard curve generated by EPRI, specifically for the McGuire site, is used as the basis for this analysis. According to GL 88-20, Supplement 4, IPEEE resolves this issue by using both the LLNL and EPRI hazard curves in evaluating the seismic risk. A sensitivity study was performed using the January 1989 Lawrence Livermore National Lab (LLNL) hazard curves for McGuire. The documentation is filed as McGuire calculation MCC-1535.00-00-0003. As shown in Table 3-5, the ranking of the dominant accident sequences are comparable with the EPRI curve results and do not add to or alter any of the insights of this analysis.

- USI A-17 "System Interactions in Nuclear Power Plants"

  The seismic review included consideration of spatial interactions due to seismic events. Potential for interaction was looked for during the plant walkdowns as discussed in Section 3.1.2. A few minor items were identified during the walkdown and appropriate action was taken. Examples are identified in Table 3-3. The walkdown did not identify any significant seismic interaction concerns. Therefore, USI A-17 is considered closed.

TABLE 3-1

COMPONENT FRAGILITIES USED IN THE MCGUIRE SEISMIC ANALYSIS

Event Name	Component / Structure	$\hat{\lambda}$ (g's)	β_r	β_u
CEQ0004DEX	Off-Site Power Line Insulators	0.30	0.25	0.50
CEQ0007DEX	Normal AFW Source Water Fails	0.40	0.30	0.52
CEQ0008DEX	Turbine-Driven CA Pump Failure	0.48	0.10	0.37
CEQ0009DEX	Charger Fails	0.49	0.33	0.32
CEQ0012DEX	DG Starting Air Tank Fails	0.68	0.29	0.40
CEQ0013DEX	Motor-Driven CA Pump Failure	0.69	0.10	0.35
CEQ0014DEX	Safety Injection Pumps Fail	0.71	0.31	0.47
CEQ0015DEX	Centrifugal Charging Pumps Fail	0.71	0.31	0.47
CEQ0016DEX	ND Pumps Fail	1.76	0.29	0.30
CEQ0017DEX	Cold Leg Accumulator Fails	0.79	0.37	0.38
CEQ0018DEX	FWST Tank Fails	0.85	0.32	0.41
CEQ0019DEX	Nuclear Service Water Piping to Pond Fails	0.92	0.49	0.31
CEQ0020DEX	Nuclear Service Water Pump Fails	0.92	0.22	0.27
CEQ0021DEX	4160V Switchgear Chatter	0.95	0.33	0.34
CEQ0022DEX	125V dc Battery / Rack Failure	0.95	0.33	0.34
CEQ0023DEX	DG Control Panel Fails	0.99	0.38	0.22
CEQ0024DEX	Containment Spray Heat Exchanger Fails	0.71	0.33	0.35
CEQ0025DEX	Instrument Air Blackout Header Fails	1.00	0.30	0.51
CEQ0026DEX	DG Battery and Chargers Fail	1.03	0.29	0.23
CEQ0027DEX	Auxiliary Building Blockwall Failure	1.70	0.15	0.52
CEQ0028DEX	ND Heat Exchanger Fails	0.61	0.31	0.33
CEQ0029DEX	Inverter Chatter	1.36	0.29	0.34
CEQ0030DEX	DG Oil Day Tank Fails	1.44	0.10	0.28
CEQ0031DEX	SSPS Cabinet Fails (includes ESFAS)	1.54	0.29	0.39
CEQ0032DEX	Component Cooling Heat Exchanger Fails	1.66	0.28	0.35
CEQ0033DEX	DC / AC Panelboard Chatter	1.66	0.32	0.45
CEQ0034DEX	600V ac Motor Control Center Chatter	1.68	0.32	0.43
CEQ0035DEX	Component Cooling Pumps Fail	1.74	0.31	0.45
CEQ0037DEX	Containment Spray Pumps Fail	1.76	0.29	0.30
CEQ0038DEX	Low Level Intake Structure Fails	0.58	0.32	0.34
CEQ0040DEX	Volume Control Tank Fails	0.94	0.37	0.51
CEQ0041DEX	Regenerative Heat Exchangers Fail	0.90	0.37	0.49
CEQ0043DEX	Containment Air Return Fans Fail	1.61	0.37	0.51
CEQ0044DEX	Auxiliary Shutdown Panel Fails	1.40	0.30	0.28

TABLE 3-1

COMPONENT FRAGILITIES USED IN THE MCGUIRE SEISMIC ANALYSIS
(Cont.)

Event Name	Component / Structure	$\hat{\lambda}$ (g's)	β_r	β_u
CEQ0045DEX	Main Control Boards Fail	0.96	0.37	0.39
CEQ0046DEX	Auxiliary Building Fails	2.10	0.24	0.39
CEQ0047DEX	Motor-Driven CA Local Control Panel Fails	1.11	0.30	0.28
CEQ0048DEX	Turbine-Driven CA Local Control Panel Fails	1.74	0.30	0.33

TABLE 3-2

FAULT TREE BASIC EVENTS USED IN THE MCGUIRE SEISMIC ANALYSIS

Event Name	Event Description	Probability
CEQ0001DHE	Failure to Recover From Relay Chatter	1.00E-01
CEQ0005DHE	Operator Fails to Align Train A to NSW Pond	3.00E-01
CEQ0042COM	Fouling of CA Intake Water From Lake or Pond Due To Seismic Activity	1.00E-01
FCA0TDPTPM	CA Turbine-Driven Pump Train in Maintenance or Testing	1.40E-02
FCA0TDPTPR	CA Turbine-Driven Pump Fails to Run	5.52E-03
FCA0TDPTPS	CA Turbine-Driven Pump Fails to Start	6.20E-03
FCAOLCPDHE	Operator Fails to Go to the Local CA Control Panel	1.00E-01
FCATHRODHE	Operator Fails to Manually Throttle the Aux. FDW Flow	1.00E-03
JDG001ADGR	Diesel Generator 1A Fails to Run	2.00E-01
JDG001ADGS	Diesel Generator 1A Fails to Start	6.00E-03
JDG001ALHE	Latent Human Error on Diesel Generator 1A	2.25E-03
JDG001ATRM	Diesel Generator 1A in Maintenance or Testing	4.36E-02
JDG001BDGR	Diesel Generator 1B Fails to Run	2.00E-01
JDG001BDGS	Diesel Generator 1B Fails to Start	6.00E-03
JDG001BLHE	Latent Human Error on Diesel Generator 1B	2.25E-03
JDG001BTRM	Diesel Generator 1B in Maintenance or Testing	4.36E-02
JDG1ARNCOM	Common Cause Failure of Diesel Generator to Run	1.39E-02
JDG1ASTCOM	Common Cause Failure of Diesel Generator To Start	1.20E-04
RVIBACKDHE	Operators Fail to Align Backup Air to PORVs	8.60E-03
TFBLD01DHE	Operators Fail to Establish Feed and Bleed Cooling	1.00E-02
TFBLD02DHE	Operators Fail to Establish Feed and Bleed With Small Leak Present	1.00E-01
TNCOSRVDEX	Pressurizer SRV Fails to Reseat After Relieving Liquid	1.00E-01
TRECIRCDHE	Operators Fail to Establish High Pressure Recirculation	1.00E-02
WRNTRNBTRM	RN Train 1B in Maintenance	2.80E-02

TABLE 3-3

ENHANCEMENTS RESULTING FROM THE IPEEE SEISMIC VERIFICATION WALKDOWN

Issue	Resolution
Gaps between end batteries and racks on Unit 1 Diesel Generator batteries	Spacers were installed.
Grout missing between Component Cooling heat exchangers saddle base and concrete curb	Grout will be installed. Problem Investigation Process has been initiated.
Grating in contact with Steam Vent valves	Grating will be trimmed to maintain clearance during upcoming outages. Work Request has been written.
Bolts missing from Unit 2 Upper Surge Tanks	Bolts were re-installed.
Motor Control Centers touching	MCC's were connected together to act as unit.
Potential seismic interaction from movable equipment	"Guidelines for Movable Equipment" will be developed and in place by 12/94.
Eight inch diameter pipe touching back top corner of Unit 2 Turbine Driven Auxiliary Feedwater Pump Control Panel	Panel will be modified to avoid seismic interaction. Problem Investigation Process has been initiated.
Corrosion on Auxiliary Feedwater Condensate Storage Tank anchor bolt nuts	Nuts were cleaned and recoated.
Arc Barriers loose inside Main Control Boards	Barriers will be tightened in upcoming outages. Problem Investigation Process has been initiated.

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

	FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL									ACCIDENT SEQUENCES
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	
7.60	0.00	0.00	3.87	14.66	23.49	19.28	15.77	13.54	9.39	CEQ0004DEX*CEQ0007DEX*CEQ0012DEX
7.56	8.65	47.01	27.55	11.20	4.05	1.06	0.34	0.12	0.03	-CEQ0008DEX*CEQ0004DEX*CDG001ADGR*CDG001BDGR
6.26	0.00	0.00	0.91	8.73	21.56	21.38	18.82	16.79	11.81	CEQ0004DEX*CEQ0008DEX*CEQ0012DEX
3.80	0.00	0.00	9.67	21.59	25.45	17.02	11.93	9.02	5.33	CEQ0005DHE*CEQ0007DEX*CEQ0038DEX*-CEQ0043DEX
3.15	0.00	0.00	30.94	38.20	21.76	6.43	1.96	0.61	0.09	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0012DEX
3.00	0.00	0.00	1.42	8.35	18.33	19.09	18.79	18.69	15.34	CEQ0004DEX*CEQ0007DEX*CEQ0022DEX
2.80	0.00	0.00	2.54	14.37	26.11	21.08	15.91	12.50	7.49	CEQ0005DHE*CEQ0008DEX*CEQ0038DEX*-CEQ0043DEX
2.67	0.00	0.00	0.31	4.60	15.55	19.56	20.73	21.42	17.84	CEQ0004DEX*CEQ0008DEX*CEQ0022DEX
2.63	8.65	47.01	27.55	11.20	4.05	1.06	0.34	0.12	0.03	-CEQ0008DEX*CEQ0004DEX*CDG1ARNCOM
2.59	0.00	0.00	46.92	34.14	14.32	3.44	0.90	0.25	0.03	-CEQ0008DEX*-CEQ0007DEX*CEQ0005DHE*CEQ0038DEX*-CEQ0043DEX
2.14	0.00	0.00	0.00	5.12	14.87	18.45	20.37	21.88	19.32	CEQ0004DEX*CEQ0007DEX*CEQ0023DEX
2.08	0.00	0.00	0.74	6.40	17.12	19.67	20.15	20.08	15.83	CEQ0007DEX*CEQ0019DEX*CEQ0038DEX*-CEQ0043DEX
2.06	0.00	0.00	4.74	16.77	24.90	19.07	14.68	11.99	7.85	CEQ0004DEX*CEQ0007DEX*CEQ0038DEX*CDG001BDGR
1.99	0.00	0.00	0.00	2.70	12.09	18.12	21.53	24.03	21.53	CEQ0004DEX*CEQ0008DEX*CEQ0023DEX
1.93	0.00	0.00	0.00	3.29	12.52	18.09	21.63	23.91	20.56	CEQ0007DEX*CEQ0020DEX*-CEQ0043DEX
1.89	0.00	0.00	0.16	3.46	14.24	19.76	21.79	22.56	18.05	CEQ0008DEX*CEQ0019DEX*CEQ0038DEX*-CEQ0043DEX
1.86	0.00	0.00	22.45	34.83	25.91	10.27	4.27	1.82	0.45	-CEQ0008DEX*CEQ0004DEX*CEQ0038DEX*CDG001BDGR
1.82	0.00	0.00	0.00	1.71	10.01	17.46	22.48	25.82	22.53	CEQ0008DEX*CEQ0020DEX*-CEQ0043DEX
1.82	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ADGR*CDG001BDGR
1.65	0.00	0.00	1.14	10.26	23.48	21.71	18.00	15.27	10.15	CEQ0004DEX*CEQ0008DEX*CEQ0038DEX*CDG001BDGR
1.65	8.65	47.01	27.55	11.20	4.05	1.06	0.34	0.12	0.03	-CEQ0008DEX*CEQ0004DEX*CDG001ATRM*CDG001BDGR
1.65	8.65	47.01	27.55	11.20	4.05	1.06	0.34	0.12	0.03	-CEQ0008DEX*CEQ0004DEX*CDG001ADGR*CDG001BTRM
1.52	0.00	0.00	0.02	0.98	7.98	16.55	23.18	27.36	23.93	CEQ0008DEX*CEQ0013DEX*CEQ0018DEX*-CEQ0043DEX
1.43	0.00	0.00	0.04	2.47	15.91	24.75	25.15	21.03	10.65	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
1.31	0.00	0.00	0.00	0.00	8.45	15.23	21.46	27.11	27.76	CEQ0004DEX*CEQ0007DEX*CEQ0026DEX
1.30	0.00	0.00	0.03	1.15	7.54	14.84	21.51	27.28	27.65	CEQ0007DEX*CEQ0009DEX*CEQ0022DEX*CEQ0038DEX
1.28	0.00	0.00	0.00	0.00	6.53	14.21	21.56	28.30	29.39	CEQ0004DEX*CEQ0008DEX*CEQ0026DEX
1.27	0.00	0.00	0.00	0.58	5.85	13.90	21.70	28.58	29.39	CEQ0008DEX*CEQ0009DEX*CEQ0022DEX*CEQ0038DEX
1.23	0.00	0.00	0.00	0.22	3.88	12.44	22.48	30.81	30.16	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*-CEQ0043DEX*CEQ0024DEX
1.00	0.00	0.00	0.00	0.17	3.13	10.59	20.59	30.75	34.77	CEQ0008DEX*CEQ0009DEX*CEQ0013DEX*CEQ0022DEX
0.92	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ADGR*CDG001BDGR
0.87	0.00	0.00	0.03	1.99	13.95	23.39	25.46	22.68	12.50	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.80	0.00	0.00	10.69	25.26	28.64	17.02	9.98	6.03	2.37	CEQ0007DEX*CEQ0028DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.79	0.00	0.00	0.00	0.17	3.19	11.02	21.33	31.13	33.16	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*-CEQ0043DEX*CEQ0024DEX
0.74	0.00	0.00	18.94	36.40	28.42	10.65	3.91	1.41	0.26	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0022DEX
0.70	0.00	0.00	13.49	20.40	21.98	15.22	11.72	10.00	7.19	CEQ0045DEX*FCAOLCPDHE
0.63	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG1ARNCOM
0.54	0.00	0.00	0.00	0.06	1.67	7.83	18.80	32.00	39.65	CEQ0008DEX*CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*-CEQ0043DEX*CEQ0024DEX
0.52	0.00	0.00	6.12	16.23	23.18	18.37	14.85	12.65	8.59	CEQ0007DEX*CEQ0018DEX*CEQ0042COM*-CEQ0043DEX
0.50	0.00	0.00	0.00	0.78	8.56	19.48	26.31	27.34	17.52	CEQ0008DEX*CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.45	0.00	0.00	4.74	16.77	24.90	19.07	14.68	11.99	7.85	CEQ0004DEX*CEQ0007DEX*CEQ0038DEX*CDG001BTRM

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.43	0.00	0.35	8.64	22.86	28.24	18.10	11.37	7.31	3.13	CEQ0004DEX*CEQ0007DEX*CEQ0017DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.42	0.00	0.00	12.21	34.32	32.63	13.49	5.16	1.87	0.33	-CEQ0008DEX*-CEQ0007DEX*CEQ0019DEX*CEQ0038DEX*-CEQ0043DEX
0.40	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ATRM*CDG001BDGR
0.40	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ADGR*CDG001BTRM
0.36	0.00	0.00	1.14	10.26	23.48	21.71	18.00	15.27	10.15	CEQ0004DEX*CEQ0008DEX*CEQ0038DEX*CDG001BTRM
0.34	0.00	0.00	7.97	30.76	34.98	16.29	6.82	2.67	0.52	-CEQ0008DEX*-CEQ0007DEX*CEQ0009DEX*CEQ0022DEX
0.34	0.00	0.00	2.94	11.07	19.79	18.61	17.36	16.76	13.46	CEQ0001DHE*CEQ0007DEX*CEQ0021DEX
0.33	0.00	0.00	0.00	36.07	37.25	16.64	6.85	2.67	0.53	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0023DEX
0.32	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG1ARNCOM
0.30	0.00	0.00	0.00	0.00	7.15	13.35	19.70	27.21	32.59	CEQ0004DEX*CEQ0012DEX*CEQ0027DEX
0.29	0.00	0.00	0.47	5.34	16.45	20.15	21.01	20.78	15.80	CEQ0007DEX*CEQ0028DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.29	0.00	0.00	0.66	6.31	17.38	19.74	19.83	19.89	16.19	CEQ0001DHE*CEQ0008DEX*CEQ0021DEX
0.26	0.00	0.00	0.00	0.03	1.16	6.21	16.69	31.42	44.49	CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0025DEX*-CEQ0043DEX*CEQ0024DEX
0.26	0.00	0.00	0.00	0.00	2.06	7.64	17.23	30.23	42.85	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0031DEX
0.24	0.00	0.00	0.00	28.43	38.42	19.97	8.91	3.58	0.69	-CEQ0008DEX*-CEQ0007DEX*CEQ0020DEX*-CEQ0043DEX
0.23	0.00	0.00	0.00	0.51	6.50	16.84	25.46	29.26	21.43	CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0025DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.22	0.00	0.00	0.38	4.45	14.24	18.42	20.67	22.28	19.56	CEQ0007DEX*CEQ0009DEX*CEQ0022DEX*CEQ0042COM
0.21	0.00	0.00	0.00	0.25	2.68	8.11	16.97	29.27	42.72	CEQ0007DEX*CEQ0020DEX*CEQ0043DEX
0.21	0.00	0.00	0.00	0.12	2.02	7.37	16.62	29.78	44.09	CEQ0008DEX*CEQ0020DEX*CEQ0043DEX
0.21	0.00	0.00	33.39	38.51	20.34	5.60	1.61	0.48	0.07	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0038DEX*CDG001BTRM
0.20	0.00	0.00	0.00	0.56	4.25	10.22	18.33	28.50	38.14	CEQ0007DEX*CEQ0019DEX*CEQ0038DEX*CEQ0043DEX
0.20	0.00	0.00	0.00	0.28	3.23	9.38	18.11	29.26	39.73	CEQ0008DEX*CEQ0019DEX*CEQ0038DEX*CEQ0043DEX
0.20	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ATRM*CDG001BDGR
0.20	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ADGR*CDG001BTRM
0.19	0.00	0.00	0.00	3.54	11.80	16.52	20.26	23.92	23.96	CEQ0005DHE*CEQ0007DEX*CEQ0038DEX*CEQ0043DEX
0.18	0.00	0.00	0.00	0.07	1.54	6.71	16.44	30.29	44.95	CEQ0008DEX*CEQ0013DEX*CEQ0018DEX*CEQ0043DEX
0.18	0.00	0.00	0.00	1.83	9.40	15.89	20.99	25.74	26.15	CEQ0005DHE*CEQ0008DEX*CEQ0038DEX*CEQ0043DEX
0.18	0.00	0.00	0.34	4.28	14.37	18.99	21.21	22.34	18.48	CEQ0004DEX*CEQ0007DEX*CEQ0017DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.17	0.00	0.00	0.00	0.00	16.35	22.10	23.40	22.72	15.43	CEQ0027DEX*CEQ0028DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.17	0.00	0.00	0.00	0.01	0.67	4.48	14.17	30.31	50.36	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*CEQ0043DEX*CEQ0024DEX
0.17	0.00	0.02	2.44	13.25	25.48	22.15	17.27	12.95	6.44	CEQ0007DEX*CEQ0014DEX*CEQ0015DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.17	0.00	0.00	10.52	21.10	24.23	16.42	11.96	9.56	6.22	CEQ0004DEX*CEQ0012DEX*FCA0TDPTPM
0.16	0.00	0.00	0.00	0.00	4.19	9.93	17.64	28.23	40.01	CEQ0004DEX*CEQ0022DEX*CEQ0027DEX
0.16	0.00	0.00	0.00	0.00	3.53	9.83	18.51	29.45	38.67	CEQ0027DEX*CEQ0028DEX*-CEQ0043DEX*CEQ0024DEX
0.14	0.00	0.00	0.00	0.00	2.78	8.19	16.60	28.78	43.64	CEQ0009DEX*CEQ0022DEX*CEQ0027DEX
0.13	0.00	0.00	0.00	0.00	2.94	8.30	16.55	28.60	43.61	CEQ0004DEX*CEQ0023DEX*CEQ0027DEX
0.13	0.00	0.00	0.00	0.23	4.14	13.48	23.97	31.29	26.89	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.13	0.00	0.00	0.00	0.00	0.39	3.24	12.04	29.28	55.04	CEQ0008DEX*CEQ0028DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.12	0.00	0.00	0.00	0.00	2.34	7.69	16.60	29.53	43.84	CEQ0020DEX*CEQ0027DEX*-CEQ0043DEX
0.12	0.00	0.00	0.00	0.00	1.17	5.92	15.91	30.73	46.27	CEQ0008DEX*CEQ0013DEX*CEQ0035DEX*-CEQ0043DEX*CEQ0024DEX
0.12	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPM*CDG001ADGR*CDG001BDGR

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

	FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL									
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.11	0.00	0.00	0.00	0.00	2.86	8.59	17.31	29.34	41.90	CEQ0004DEX*CEQ0017DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0024DEX
0.11	0.00	0.00	0.00	0.00	3.74	9.77	18.07	28.97	39.45	CEQ0019DEX*CEQ0027DEX*CEQ0038DEX*-CEQ0043DEX
0.11	0.00	0.00	0.00	0.00	0.00	0.00	11.41	27.71	60.88	CEQ0004DEX*CEQ0007DEX*CEQ0030DEX
0.11	0.00	0.00	0.00	0.00	0.53	3.82	12.93	29.46	53.25	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0043DEX*CEQ0024DEX
0.11	0.00	0.00	0.00	0.00	0.30	2.58	10.33	27.37	59.42	CEQ0008DEX*CEQ0009DEX*CEQ0022DEX*CEQ0045DEX*CEQ0047DEX
0.11	0.00	0.00	0.00	0.00	0.00	0.00	10.93	27.58	61.48	CEQ0004DEX*CEQ0008DEX*CEQ0030DEX
0.11	0.00	0.00	0.00	0.00	10.90	16.58	20.98	25.53	26.02	CEQ0005DHE*CEQ0027DEX*CEQ0038DEX*-CEQ0043DEX
0.11	5.21	33.99	27.33	15.97	9.16	4.03	2.18	1.40	0.72	CEQ0007DEX*CEQ0042COM*TFBLD01DHE*-CEQ0043DEX
0.11	0.00	0.00	29.40	36.08	22.94	7.76	2.70	0.95	0.17	-CEQ0008DEX*-CEQ0007DEX*CEQ0001DHE*CEQ0021DEX
0.10	0.00	0.00	0.00	0.00	14.12	20.58	23.33	24.14	17.83	CEQ0004DEX*CEQ0017DEX*CEQ0027DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.10	0.00	0.00	0.00	0.00	1.35	5.53	14.06	28.57	50.50	CEQ0004DEX*CEQ0026DEX*CEQ0027DEX
0.10	0.00	0.00	0.00	0.00	2.55	7.99	16.75	29.31	43.39	CEQ0018DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0024DEX
0.10	0.00	0.00	0.00	0.00	6.69	16.42	24.82	29.28	22.80	CEQ0008DEX*CEQ0013DEX*CEQ0035DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.10	0.00	0.00	0.06	1.67	8.72	15.63	21.67	26.62	25.62	CEQ0007DEX*CEQ0014DEX*CEQ0015DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.10	0.00	0.00	22.45	26.54	22.44	12.38	7.73	5.44	3.02	CEQ0005DHE*CEQ0038DEX*FCA0TDPTPM*-CEQ0043DEX
0.10	5.58	36.40	28.81	15.89	8.12	3.01	1.32	0.66	0.22	CEQ0007DEX*CEQ0042COM*TRECIRCDHE*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.09	0.00	0.00	0.00	0.00	13.01	19.77	23.29	24.88	19.05	CEQ0018DEX*CEQ0027DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.09	0.00	0.00	0.00	0.00	0.00	0.00	22.88	33.06	44.06	CEQ0046DEX
0.09	0.00	0.00	0.00	0.00	0.31	2.74	10.91	28.25	57.79	CEQ0004DEX*CEQ0008DEX*CEQ0017DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.09	0.00	0.00	0.00	0.11	2.63	10.56	22.07	32.77	31.86	CEQ0008DEX*CEQ0028DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.09	0.00	0.00	0.00	0.00	45.83	29.74	15.63	7.17	1.64	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0026DEX
0.08	0.00	0.00	0.00	0.00	0.25	2.50	10.52	27.95	58.76	CEQ0008DEX*CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*CEQ0043DEX*CEQ0024DEX
0.08	0.00	0.00	0.00	0.17	3.42	12.01	22.87	31.80	29.73	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.08	0.00	0.00	0.00	0.00	0.27	2.52	10.40	27.82	58.98	CEQ0008DEX*CEQ0018DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.08	0.00	0.00	0.00	0.00	0.00	3.88	12.93	29.55	53.64	CEQ0008DEX*CEQ0013DEX*CEQ0032DEX*-CEQ0043DEX*CEQ0024DEX
0.08	0.00	0.00	10.52	21.10	24.23	16.42	11.96	9.55	6.22	CEQ0004DEX*CEQ0012DEX*FCA0TDPTPS
0.07	0.00	0.00	0.00	0.00	1.45	5.91	14.80	29.25	48.59	CEQ0014DEX*CEQ0015DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0024DEX
0.07	0.00	0.00	0.00	0.00	0.15	1.77	8.75	26.44	62.89	CEQ0008DEX*CEQ0014DEX*CEQ0015DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.07	0.00	0.00	0.00	0.00	8.38	14.59	20.27	26.63	30.13	CEQ0004DEX*CEQ0027DEX*CEQ0038DEX*CDG001BDGR
0.07	0.00	0.00	1.56	9.91	22.27	22.03	19.22	15.94	9.07	CEQ0007DEX*CEQ0017DEX*CEQ0025DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.07	0.00	0.00	0.00	0.00	7.69	14.63	20.96	27.26	29.45	CEQ0001DHE*CEQ0004DEX*CEQ0008DEX*CEQ0029DEX
0.07	0.00	0.00	0.00	0.00	9.91	15.62	20.77	26.01	27.70	CEQ0001DHE*CEQ0004DEX*CEQ0007DEX*CEQ0029DEX
0.07	0.00	0.00	10.52	21.10	24.23	16.42	11.96	9.55	6.22	CEQ0004DEX*CEQ0012DEX*FCA0TDPTPR
0.06	0.00	0.00	0.00	0.00	0.00	4.31	12.39	27.72	55.57	CEQ0004DEX*CEQ0012DEX*CEQ0045DEX*CEQ0048DEX
0.06	0.00	0.00	0.00	0.00	8.33	16.47	23.19	27.97	24.04	CEQ0014DEX*CEQ0015DEX*CEQ0027DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.06	0.00	0.00	0.00	0.06	1.81	8.63	20.41	33.09	35.99	CEQ0008DEX*CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.06	0.00	0.00	0.00	0.08	2.14	9.29	20.80	32.89	34.79	CEQ0004DEX*CEQ0008DEX*CEQ0017DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.06	0.00	0.00	0.00	0.00	0.00	3.13	11.76	28.92	56.19	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*-CEQ0043DEX*CEQ0037DEX
0.06	0.00	0.00	0.00	0.07	1.92	8.68	20.20	32.97	36.16	CEQ0008DEX*CEQ0018DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.06	0.00	0.00	0.00	0.00	0.00	12.58	23.59	32.92	30.91	CEQ0008DEX*CEQ0013DEX*CEQ0032DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.06	0.00	0.00	4.34	13.55	21.33	18.34	16.08	14.88	11.47	CEQ0004DEX*CEQ0022DEX*FCA0TDPTPM

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.06	0.00	0.00	4.74	16.77	24.90	19.07	14.68	11.99	7.85	CEQ0004DEX*CEQ0007DEX*CEQ0038DEX*CDG001BDGS
0.05	0.00	0.00	0.00	0.00	0.10	1.51	8.26	26.15	63.98	CEQ0008DEX*CEQ0013DEX*CEQ0044DEX*CEQ0045DEX*-CEQ0043DEX*CEQ0024DEX
0.05	0.00	0.00	0.00	0.00	0.00	2.86	11.10	28.35	57.69	CEQ0008DEX*CEQ0013DEX*CEQ0016DEX*-CEQ0043DEX*CEQ0024DEX
0.05	0.00	0.00	0.00	0.00	10.89	15.45	20.13	25.48	28.05	CEQ0004DEX*CEQ0007DEX*CEQ0031DEX*CEQ0042COM
0.05	0.00	0.00	0.03	1.06	6.50	13.24	20.55	27.92	30.70	CEQ0007DEX*CEQ0017DEX*CEQ0025DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.05	0.00	0.00	0.00	0.00	4.36	11.29	19.70	29.17	35.49	CEQ0001DHE*CEQ0008DEX*CEQ0029DEX*CEQ0038DEX
0.05	0.00	0.00	0.00	0.00	5.71	12.24	19.84	28.29	33.92	CEQ0001DHE*CEQ0007DEX*CEQ0029DEX*CEQ0038DEX
0.05	0.00	0.00	0.00	0.00	9.74	15.91	21.01	26.09	27.26	CEQ0001DHE*CEQ0004DEX*CEQ0008DEX*CEQ0033DEX
0.05	0.00	0.00	0.00	0.00	12.44	16.83	20.64	24.68	25.41	CEQ0001DHE*CEQ0004DEX*CEQ0007DEX*CEQ0033DEX
0.05	0.00	0.00	1.14	10.26	23.48	21.71	18.00	15.27	10.15	CEQ0004DEX*CEQ0008DEX*CEQ0038DEX*CDG001BDGS
0.05	0.00	0.00	12.42	23.26	24.77	15.66	10.73	8.15	5.02	CEQ0004DEX*CEQ0038DEX*FCA0TDPTPM*CDG001BDGR
0.05	0.00	0.00	22.45	26.54	22.44	12.38	7.73	5.44	3.02	CEQ0005DHE*CEQ0038DEX*FCA0TDPTPS*-CEQ0043DEX
0.05	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ADGS*CDG001BDGR
0.05	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ADGR*CDG001BDGS
0.05	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPS*CDG001ADGR*CDG001BDGR
0.05	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPR*CDG001ADGR*CDG001BDGR
0.04	0.00	0.00	0.00	0.00	0.99	4.59	12.87	28.13	53.41	CEQ0017DEX*CEQ0025DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0024DEX
0.04	0.00	0.00	0.00	0.03	1.11	6.50	18.07	33.31	40.99	CEQ0008DEX*CEQ0014DEX*CEQ0015DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.04	0.00	0.00	0.00	0.00	0.17	1.89	8.91	26.16	62.87	CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0025DEX*CEQ0043DEX*CEQ0024DEX
0.04	0.00	0.00	0.00	0.00	0.00	2.65	10.63	27.85	58.87	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*-CEQ0043DEX*CEQ0037DEX
0.04	0.00	0.00	2.36	10.77	20.63	19.58	17.85	16.55	12.26	CEQ0019DEX*CEQ0038DEX*FCA0TDPTPM*-CEQ0043DEX
0.04	0.00	0.00	0.00	8.97	18.68	19.14	18.81	18.80	15.59	CEQ0004DEX*CEQ0023DEX*FCA0TDPTPM
0.04	0.00	0.00	0.00	5.93	16.15	19.26	20.51	21.10	17.04	CEQ0020DEX*FCA0TDPTPM*-CEQ0043DEX
0.04	0.00	0.00	0.00	0.00	52.60	26.86	13.06	6.00	1.47	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0031DEX
0.04	0.00	0.00	0.00	0.00	2.26	8.35	18.15	30.48	40.76	CEQ0001DHE*CEQ0008DEX*CEQ0013DEX*CEQ0029DEX
0.04	0.00	0.00	0.00	0.00	7.31	13.46	[illegible]	27.38	31.74	CEQ0001DHE*CEQ0007DEX*CEQ0033DEX*CEQ0038DEX
0.04	0.00	0.00	0.00	0.00	5.61	12.48	[illegible]	28.41	33.41	CEQ0001DHE*CEQ0008DEX*CEQ0033DEX*CEQ0038DEX
0.04	0.00	0.00	22.45	26.54	22.44	12.38	7.73	5.44	3.02	CEQ0005DHE*CEQ0038DEX*FCA0TDPTPR*-CEQ0043DEX
0.04	0.08	10.27	27.45	25.22	17.95	8.81	5.07	3.37	1.77	CEQ0004DEX*CEQ0007DEX*CEQ0042COM*RVIBACKDHE*-CEQ0043DEX
0.04	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPM*CDG1ARNCOM
0.03	0.00	0.00	0.00	0.00	0.00	2.30	8.93	24.99	63.78	CEQ0004DEX*CEQ0023DEX*CEQ0045DEX*CEQ0048DEX
0.03	0.00	0.00	0.00	0.00	0.00	2.88	9.97	25.85	61.30	CEQ0004DEX*CEQ0022DEX*CEQ0045DEX*CEQ0048DEX
0.03	0.00	0.00	0.00	0.00	0.00	2.11	8.86	25.55	63.48	CEQ0020DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX
0.03	0.00	0.00	0.00	0.00	6.19	13.92	21.92	29.25	28.73	CEQ0017DEX*CEQ0025DEX*CEQ0027DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	0.10	1.33	7.34	24.53	66.70	CEQ0008DEX*CEQ0017DEX*CEQ0025DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	0.00	1.70	8.50	25.96	63.83	CEQ0008DEX*CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*-CEQ0043DEX*CEQ0037DEX
0.03	0.00	0.00	0.00	0.04	1.27	6.91	18.28	32.77	40.73	CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0025DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	0.78	5.64	17.41	33.62	42.55	CEQ0008DEX*CEQ0013DEX*CEQ0044DEX*CEQ0045DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	0.17	1.68	7.80	24.28	66.07	CEQ0004DEX*CEQ0008DEX*CEQ0031DEX*CEQ0045DEX*CEQ0047DEX
0.03	0.00	0.00	0.00	0.00	0.00	9.84	21.46	33.47	35.23	CEQ0008DEX*CEQ0013DEX*CEQ0016DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	2.07	8.35	13.86	19.61	26.06	30.05	CEQ0007DEX*CEQ0018DEX*CEQ0042COM*CEQ0043DEX

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.03	0.00	0.00	0.00	5.35	17.12	21.31	21.87	20.60	13.75	CEQ0007DEX*CEQ0028DEX*CEQ0042COM*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.46	4.02	10.30	18.80	29.00	37.42	CEQ0007DEX*CEQ0028DEX*CEQ0042COM*CEQ0043DEX*CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	25.81	24.21	21.12	17.98	10.87	CEQ0007DEX*CEQ0035DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.65	7.75	23.24	24.72	20.77	15.56	7.30	CEQ0013DEX*CEQ0028DEX*FCA0TDPTPM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	2.95	9.36	18.75	30.06	38.88	CEQ0001DHE*CEQ0008DEX*CEQ0013DEX*CEQ0033DEX
0.03	0.00	0.00	0.68	9.05	26.27	25.48	19.33	13.42	5.77	CEQ0008DEX*CEQ0013DEX*TRECIRCDHE*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	7.71	13.64	19.99	27.08	31.58	CEQ0001DHE*CEQ0007DEX*CEQ0034DEX*CEQ0038DEX
0.03	0.00	0.00	0.68	9.05	26.27	25.48	19.33	13.42	5.77	CEQ0008DEX*CEQ0013DEX*TFBLD01DHE*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.03	0.00	0.00	0.00	0.00	5.93	12.67	19.99	28.13	33.29	CEQ0001DHE*CEQ0008DEX*CEQ0034DEX*CEQ0038DEX
0.03	0.00	0.00	0.00	0.00	3.12	9.52	18.70	29.84	38.82	CEQ0001DHE*CEQ0008DEX*CEQ0013DEX*CEQ0034DEX
0.03	0.00	0.00	4.34	13.55	21.33	18.34	16.08	14.88	11.47	CEQ0004DEX*CEQ0022DEX*FCA0TDPTPS
0.03	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ADGS*CDG001BDGR
0.03	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ADGR*CDG001BDGS
0.03	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPM*CDG001ATRM*CDG001BDGR
0.03	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPM*CDG001ADGR*CDG001BTRM
0.03	0.00	0.00	33.82	38.67	20.12	5.41	1.50	0.43	0.06	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0038DEX*CDG001BDGS*-CEQ0043DEX
0.02	0.00	0.00	0.00	0.00	0.43	2.73	9.63	25.47	61.73	CEQ0004DEX*CEQ0017DEX*CEQ0027DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.60	3.27	10.68	26.57	58.87	CEQ0004DEX*CEQ0027DEX*CEQ0031DEX*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.00	0.99	6.28	22.99	69.74	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.38	2.50	9.17	25.04	62.91	CEQ0018DEX*CEQ0027DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.00	1.20	7.04	24.22	67.54	CEQ0013DEX*CEQ0028DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.99	4.35	12.18	27.32	55.16	CEQ0027DEX*CEQ0035DEX*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.00	1.38	7.05	23.30	68.26	CEQ0009DEX*CEQ0022DEX*CEQ0038DEX*CEQ0045DEX*CEQ0048DEX
0.02	0.00	0.00	0.00	0.00	0.00	1.42	7.02	23.13	68.43	CEQ0004DEX*CEQ0026DEX*CEQ0045DEX*CEQ0048DEX
0.02	0.00	0.00	0.00	0.00	0.00	0.93	5.89	22.08	71.10	CEQ0009DEX*CEQ0013DEX*CEQ0022DEX*CEQ0045DEX*CEQ0048DEX
0.02	0.00	0.00	0.00	0.00	3.82	10.81	20.03	30.36	34.98	CEQ0027DEX*CEQ0028DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.56	3.24	10.67	26.50	59.03	CEQ0027DEX*CEQ0028DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.59	3.20	10.36	25.93	59.91	CEQ0019DEX*CEQ0027DEX*CEQ0038DEX*CEQ0043DEX
0.02	0.00	0.00	0.00	0.00	0.35	2.39	9.03	25.07	63.16	CEQ0020DEX*CEQ0027DEX*CEQ0043DEX
0.02	0.00	0.00	0.00	0.00	0.00	2.84	10.21	26.52	60.43	CEQ0019DEX*CEQ0038DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX
0.02	0.00	0.00	0.00	0.02	0.77	5.12	15.93	32.49	45.68	CEQ0008DEX*CEQ0017DEX*CEQ0025DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.04	0.75	5.20	21.03	72.98	CEQ0004DEX*CEQ0008DEX*CEQ0017DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.00	1.27	7.11	24.05	67.56	CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0025DEX*-CEQ0043DEX*CEQ0037DEX
0.02	0.00	0.00	0.00	0.00	0.05	0.90	5.86	22.24	70.95	CEQ0008DEX*CEQ0028DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.03	0.68	4.92	20.52	73.84	CEQ0008DEX*CEQ0018DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.17	1.78	8.37	25.23	64.46	CEQ0008DEX*CEQ0013DEX*CEQ0035DEX*CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.10	1.24	6.88	23.59	68.19	CEQ0008DEX*CEQ0035DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.03	0.71	5.20	21.39	72.67	CEQ0008DEX*CEQ0044DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	0.00	6.03	14.83	29.25	49.89	CEQ0005DHE*CEQ0038DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX
0.02	0.00	0.00	0.00	4.31	15.02	20.15	22.16	22.23	16.14	CEQ0004DEX*CEQ0007DEX*CEQ0017DEX*CEQ0042COM*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.02	0.00	0.00	0.00	0.34	3.26	9.03	17.65	29.00	40.72	CEQ0004DEX*CEQ0007DEX*CEQ0017DEX*CEQ0042COM*CEQ0043DEX*CEQ0024DEX

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.02	0.00	0.00	0.00	0.00	6.66	12.89	19.99	27.87	32.59	CEQ0007DEX*CEQ0035DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	8.38	14.59	20.27	26.63	30.13	CEQ0004DEX*CEQ0027DEX*CEQ0038DEX*CDG001BTRM
0.02	0.00	0.00	0.00	0.00	16.39	18.77	20.19	22.55	22.10	CEQ0004DEX*CEQ0027DEX*CDG001ADGR*CDG001BDGR
0.02	0.00	0.00	0.01	0.68	5.76	13.35	21.45	28.70	30.04	CEQ0009DEX*CEQ0013DEX*CEQ0022DEX*FCA0TDPTPM
0.02	0.00	0.00	0.09	2.17	10.23	16.62	21.46	25.32	24.10	CEQ0009DEX*CEQ0022DEX*CEQ0038DEX*FCA0TDPTPM
0.02	0.00	0.00	0.02	0.86	7.01	15.38	22.98	28.20	25.56	CEQ0013DEX*CEQ0028DEX*FCA0TDPTPM*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	11.54	17.19	21.56	25.34	24.37	CEQ0004DEX*CEQ0026DEX*FCA0TDPTPM
0.02	0.00	0.00	0.47	5.90	19.83	23.71	22.17	18.26	9.66	CEQ0013DEX*CEQ0018DEX*FCA0TDPTPM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.02	0.00	0.00	0.49	6.44	21.04	24.14	21.72	17.33	8.84	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*FCA0TDPTPM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.02	0.00	0.00	0.00	0.00	4.98	10.64	17.93	27.85	38.60	CEQ0001DHE*CEQ0021DEX*CEQ0027DEX
0.02	0.00	0.00	0.02	1.11	8.74	17.47	23.57	26.81	22.28	CEQ0008DEX*CEQ0013DEX*TRECIRCDHE*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.02	1.11	8.74	17.47	23.57	26.81	22.28	CEQ0008DEX*CEQ0013DEX*TFBLD01DHE*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	0.32	6.69	24.11	26.12	21.02	15.08	6.65	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*RVIBACKDHE*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.02	0.00	0.00	0.00	0.76	7.41	16.56	23.69	27.85	23.73	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*RVIBACKDHE*-CEQ0043DEX*CEQ0024DEX
0.02	0.00	0.00	2.36	10.77	20.63	19.58	17.85	16.55	12.26	CEQ0019DEX*CEQ0038DEX*FCA0TDPTPS*-CEQ0043DEX
0.02	0.00	0.00	0.00	8.97	18.68	19.14	18.81	18.80	15.59	CEQ0004DEX*CEQ0023DEX*FCA0TDPTPS
0.02	0.00	0.00	0.00	5.93	16.15	19.26	20.51	21.10	17.04	CEQ0020DEX*FCA0TDPTPS*-CEQ0043DEX
0.02	0.00	0.00	0.00	8.97	18.68	19.14	18.81	18.80	15.59	CEQ0004DEX*CEQ0023DEX*FCA0TDPTPR
0.02	0.00	0.00	4.34	13.55	21.33	18.34	16.08	14.88	11.47	CEQ0004DEX*CEQ0022DEX*FCA0TDPTPR
0.02	0.00	0.00	2.36	10.77	20.63	19.58	17.85	16.55	12.26	CEQ0019DEX*CEQ0038DEX*FCA0TDPTPR*-CEQ0043DEX
0.02	0.00	0.00	0.00	31.28	37.04	18.66	8.54	3.66	0.82	-CEQ0008DEX*-CEQ0007DEX*CEQ0005DHE*CEQ0038DEX*CEQ0043DEX
0.02	0.00	0.00	1.14	10.26	23.48	21.71	18.00	15.27	10.15	CEQ0004DEX*CEQ0008DEX*CEQ0038DEX*CDG001BLHE
0.02	0.00	0.00	4.74	16.77	24.90	19.07	14.68	11.99	7.85	CEQ0004DEX*CEQ0007DEX*CEQ0038DEX*CDG001BLHE
0.02	0.00	0.00	12.42	23.26	24.77	15.66	10.73	8.15	5.02	CEQ0004DEX*CEQ0038DEX*FCA0TDPTPS*CDG001BDGR
0.02	0.00	0.00	12.42	23.26	24.77	15.66	10.73	8.15	5.02	CEQ0004DEX*CEQ0038DEX*FCA0TDPTPR*CDG001BDGR
0.02	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ALHE*CDG001BDGR
0.02	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ADGR*CDG001BLHE
0.02	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPS*CDG1ARNCOM
0.02	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPR*CDG1ARNCOM
0.01	0.00	0.00	0.00	0.00	0.00	1.85	7.75	23.58	66.82	CEQ0004DEX*CEQ0017DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	3.11	9.49	18.84	30.42	38.13	CEQ0004DEX*CEQ0017DEX*CEQ0027DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	4.10	10.88	19.96	30.32	34.74	CEQ0004DEX*CEQ0027DEX*CEQ0031DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.46	4.02	18.98	76.55	CEQ0013DEX*CEQ0017DEX*CEQ0025DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	9.90	19.34	31.33	39.44	CEQ0027DEX*CEQ0032DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	2.77	9.61	25.51	62.10	CEQ0027DEX*CEQ0032DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	4.01	14.23	31.82	49.94	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.42	3.60	17.54	78.44	CEQ0009DEX*CEQ0022DEX*CEQ0045DEX*CEQ0047DEX*CEQ0048DEX
0.01	0.00	0.00	0.00	0.00	0.00	1.69	7.35	23.10	67.86	CEQ0018DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	2.79	8.86	18.28	30.47	39.60	CEQ0018DEX*CEQ0027DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	3.70	13.63	31.47	51.20	CEQ0013DEX*CEQ0018DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.91	5.94	22.47	70.68	CEQ0013DEX*CEQ0018DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.01	0.00	0.00	0.00	0.00	0.00	4.69	15.55	32.65	47.10	CEQ0013DEX*CEQ0028DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	6.29	13.42	21.14	28.94	30.22	CEQ0027DEX*CEQ0035DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.55	4.07	18.28	77.10	CEQ0020DEX*CEQ0045DEX*CEQ0048DEX*CEQ0043DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.55	4.37	19.51	75.57	CEQ0028DEX*CEQ0045DEX*CEQ0047DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.08	1.03	5.91	21.72	71.25	CEQ0027DEX*CEQ0044DEX*CEQ0045DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	2.21	8.66	24.72	64.41	CEQ0027DEX*CEQ0028DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.41	3.64	17.88	78.07	CEQ0018DEX*CEQ0045DEX*CEQ0047DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.13	1.31	6.40	21.83	70.33	CEQ0017DEX*CEQ0025DEX*CEQ0027DEX*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	1.60	6.62	16.31	30.70	44.77	CEQ0014DEX*CEQ0015DEX*CEQ0027DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.21	1.75	7.67	23.66	66.71	CEQ0014DEX*CEQ0015DEX*CEQ0027DEX*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.00	5.07	19.80	75.13	CEQ0004DEX*CEQ0027DEX*CEQ0030DEX
0.01	0.00	0.00	0.00	0.00	0.00	2.01	8.13	24.10	65.76	CEQ0016DEX*CEQ0027DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	2.64	11.64	30.34	55.38	CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.62	4.88	20.86	73.63	CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*CEQ0045DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.46	3.85	18.36	77.33	CEQ0004DEX*CEQ0017DEX*CEQ0045DEX*CEQ0047DEX*CEQ0048DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.01	0.34	3.29	17.16	79.19	CEQ0008DEX*CEQ0017DEX*CEQ0025DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.44	3.88	18.50	77.18	CEQ0008DEX*CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.17	2.03	9.89	28.47	59.44	CEQ0008DEX*CEQ0014DEX*CEQ0015DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.02	0.47	4.02	18.96	76.53	CEQ0008DEX*CEQ0014DEX*CEQ0015DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.49	4.06	18.87	76.58	CEQ0004DEX*CEQ0008DEX*CEQ0017DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.34	3.12	12.22	30.16	54.15	CEQ0004DEX*CEQ0008DEX*CEQ0017DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.86	5.67	21.75	71.72	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.71	5.03	20.55	73.71	CEQ0004DEX*CEQ0008DEX*CEQ0013DEX*CEQ0017DEX*CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.11	1.73	9.36	28.21	60.58	CEQ0008DEX*CEQ0013DEX*CEQ0044DEX*CEQ0045DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.01	0.39	3.77	18.60	77.23	CEQ0008DEX*CEQ0013DEX*CEQ0044DEX*CEQ0045DEX*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.60	4.59	20.04	74.77	CEQ0008DEX*CEQ0028DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.44	3.67	13.43	31.11	51.34	CEQ0008DEX*CEQ0028DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.45	3.83	18.38	77.34	CEQ0008DEX*CEQ0018DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.30	2.87	11.68	29.76	55.38	CEQ0008DEX*CEQ0018DEX*CEQ0045DEX*CEQ0047DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.72	4.99	20.49	73.80	CEQ0008DEX*CEQ0013DEX*CEQ0032DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.00	4.38	14.39	31.32	49.91	CEQ0008DEX*CEQ0013DEX*CEQ0032DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	1.09	6.36	22.68	69.87	CEQ0008DEX*CEQ0013DEX*CEQ0032DEX*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.54	4.28	19.40	75.79	CEQ0008DEX*CEQ0016DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	3.16	12.20	30.20	54.44	CEQ0008DEX*CEQ0032DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.75	5.17	20.97	73.10	CEQ0008DEX*CEQ0032DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	1.19	6.66	23.11	69.03	CEQ0008DEX*CEQ0013DEX*CEQ0035DEX*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	1.28	6.60	17.47	32.15	42.49	CEQ0008DEX*CEQ0013DEX*CEQ0035DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.77	4.88	15.16	31.74	47.45	CEQ0008DEX*CEQ0035DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.30	2.96	12.21	30.66	53.87	CEQ0008DEX*CEQ0044DEX*CEQ0045DEX*CEQ0047DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	3.26	12.44	30.26	54.04	CEQ0008DEX*CEQ0013DEX*CEQ0016DEX*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	0.78	5.29	21.09	72.83	CEQ0008DEX*CEQ0013DEX*CEQ0016DEX*CEQ0043DEX*CEQ0024DEX

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.01	0.00	0.00	0.00	0.00	2.09	6.66	14.76	28.03	48.46	CEQ0005DHE*CEQ0027DEX*CEQ0038DEX*CEQ0043DEX
0.01	0.00	0.00	0.00	0.00	0.00	4.91	13.28	28.27	53.53	CEQ0004DEX*CEQ0038DEX*CEQ0045DEX*CEQ0048DEX*CDG001BDGR
0.01	0.00	0.00	0.00	1.70	9.22	16.78	22.90	26.79	22.62	CEQ0007DEX*CEQ0014DEX*CEQ0015DEX*CEQ0042COM*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.11	1.67	6.27	15.21	29.15	47.59	CEQ0007DEX*CEQ0014DEX*CEQ0015DEX*CEQ0042COM*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	7.81	16.91	30.01	45.28	CEQ0007DEX*CEQ0028DEX*CEQ0042COM*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	0.00	20.69	26.04	29.31	23.96	CEQ0007DEX*CEQ0016DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	7.20	16.12	29.72	46.96	CEQ0007DEX*CEQ0016DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	6.71	15.56	29.42	48.30	CEQ0004DEX*CEQ0007DEX*CEQ0017DEX*CEQ0042COM*-CEQ0043DEX*CEQ0037DEX
0.01	0.00	0.00	0.00	0.00	2.29	7.46	16.47	29.45	44.33	CEQ0007DEX*CEQ0042COM*CEQ0044DEX*CEQ0045DEX*-CEQ0043DEX
0.01	0.00	0.00	0.00	0.00	0.00	25.22	27.28	27.47	20.03	CEQ0007DEX*CEQ0032DEX*CEQ0042COM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	0.00	9.46	18.20	30.02	42.32	CEQ0007DEX*CEQ0032DEX*CEQ0042COM*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	1.09	6.93	14.35	21.91	28.35	27.37	CEQ0007DEX*CEQ0017DEX*CEQ0025DEX*CEQ0042COM*CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.07	1.14	4.89	13.27	28.13	52.50	CEQ0007DEX*CEQ0017DEX*CEQ0025DEX*CEQ0042COM*CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.10	2.78	14.13	21.99	24.57	22.86	13.56	CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*FCA0TDPTPM*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.23	3.17	10.18	20.23	30.83	35.36	CEQ0013DEX*CEQ0014DEX*CEQ0015DEX*FCA0TDPTPM*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.58	5.30	13.07	21.73	29.33	29.98	CEQ0013DEX*CEQ0018DEX*FCA0TDPTPM*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.01	0.66	5.85	13.84	22.15	28.95	28.55	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*FCA0TDPTPM*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	16.39	18.77	20.19	22.55	22.10	CEQ0004DEX*CEQ0027DEX*CDG1ARNCOM
0.01	0.00	0.00	0.00	6.71	25.59	27.87	21.76	13.63	4.44	-CEQ0008DEX*-CEQ0007DEX*CEQ0020DEX*CEQ0043DEX
0.01	0.00	0.00	0.00	0.00	54.13	28.22	12.26	4.62	0.76	-CEQ0008DEX*-CEQ0007DEX*CEQ0015DEX*CEQ0035DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	11.49	30.84	26.71	17.86	10.09	3.01	-CEQ0008DEX*-CEQ0007DEX*CEQ0019DEX*CEQ0038DEX*CEQ0043DEX
0.01	0.00	0.00	0.00	0.00	1.96	6.29	14.23	27.75	49.77	CEQ0001DHE*CEQ0027DEX*CEQ0029DEX
0.01	0.00	0.00	0.00	0.00	16.39	18.77	20.19	22.55	22.10	CEQ0004DEX*CEQ0027DEX*RVIBACKDHE
0.01	0.00	0.00	0.00	0.00	16.39	18.77	20.19	22.55	22.10	CEQ0004DEX*CEQ0027DEX*CDG001ADGR*CDG001BTRM
0.01	0.00	0.00	0.00	0.00	16.39	18.77	20.19	22.55	22.10	CEQ0004DEX*CEQ0027DEX*CDG001ATRM*CDG001BDGR
0.01	0.00	0.00	0.04	2.05	13.44	22.48	25.12	23.25	13.62	CEQ0008DEX*CEQ0013DEX*CEQ0025DEX*RVIBACKDHE*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	11.54	17.19	21.56	25.34	24.37	CEQ0004DEX*CEQ0026DEX*FCA0TDPTPS
0.01	0.00	0.00	0.09	2.17	10.23	16.62	21.46	25.32	24.10	CEQ0009DEX*CEQ0022DEX*CEQ0038DEX*FCA0TDPTPS
0.01	0.00	0.00	0.49	6.44	21.04	24.14	21.72	17.33	8.84	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*FCA0TDPTPS*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.01	0.66	5.85	13.84	22.15	28.95	28.55	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*FCA0TDPTPS*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.47	5.90	19.83	23.71	22.17	18.26	9.66	CEQ0013DEX*CEQ0018DEX*FCA0TDPTPS*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.00	0.58	5.30	13.07	21.73	29.33	29.98	CEQ0013DEX*CEQ0018DEX*FCA0TDPTPS*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.01	0.68	5.76	13.35	21.45	28.70	30.04	CEQ0009DEX*CEQ0013DEX*CEQ0022DEX*FCA0TDPTPS
0.01	0.00	0.00	0.65	7.75	23.24	24.72	20.77	15.56	7.30	CEQ0013DEX*CEQ0028DEX*FCA0TDPTPS*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.02	0.86	7.01	15.38	22.98	28.20	25.56	CEQ0013DEX*CEQ0028DEX*FCA0TDPTPS*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.49	6.44	21.04	24.14	21.72	17.33	8.84	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*FCA0TDPTPR*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.01	0.66	5.85	13.84	22.15	28.95	28.55	CEQ0004DEX*CEQ0013DEX*CEQ0017DEX*FCA0TDPTPR*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.65	7.75	23.24	24.72	20.77	15.56	7.30	CEQ0013DEX*CEQ0028DEX*FCA0TDPTPR*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
0.01	0.00	0.00	0.02	0.86	7.01	15.38	22.98	28.20	25.56	CEQ0013DEX*CEQ0028DEX*FCA0TDPTPR*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	0.00	0.00	11.54	17.19	21.56	25.34	24.37	CEQ0004DEX*CEQ0026DEX*FCA0TDPTPR
0.01	0.00	0.00	0.47	5.90	19.83	23.71	22.17	18.26	9.66	CEQ0013DEX*CEQ0018DEX*FCA0TDPTPR*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX

TABLE 3-4 DOMINANT SEISMIC EVENT SEQUENCES

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	ACCIDENT SEQUENCES
0.01	0.00	0.00	0.09	2.17	10.23	16.62	21.46	25.32	24.10	CEQ0009DEX*CEQ0022DEX*CEQ0038DEX*FCA0TDPTPR
0.01	0.00	0.00	0.01	0.68	5.76	13.35	21.45	28.70	30.04	CEQ0009DEX*CEQ0013DEX*CEQ0022DEX*FCA0TDPTPR
0.01	0.00	0.00	0.00	5.93	16.15	19.26	20.51	21.10	17.04	CEQ0020DEX*FCA0TDPTPR*-CEQ0043DEX
0.01	0.00	0.00	10.52	21.10	24.23	16.42	11.96	9.55	6.22	CEQ0004DEX*CEQ0012DEX*FCATHRODHE
0.01	0.00	0.00	0.00	0.00	14.58	17.66	20.41	23.62	23.73	CEQ0001DHE*CEQ0007DEX*CEQ0033DEX*CEQ0042COM
0.01	0.00	0.00	0.00	0.00	15.29	17.80	20.18	23.24	23.48	CEQ0001DHE*CEQ0007DEX*CEQ0034DEX*CEQ0042COM
0.01	0.00	0.00	0.00	0.00	11.69	16.50	20.69	25.07	26.04	CEQ0001DHE*CEQ0007DEX*CEQ0029DEX*CEQ0042COM
0.01	0.00	0.00	6.53	17.35	24.08	18.39	14.31	11.75	7.58	CEQ0007DEX*CEQ0042COM*TRECIRCDHE*-CEQ0043DEX*CEQ0024DEX
0.01	0.00	0.00	12.42	23.26	24.77	15.66	10.73	8.15	5.02	CEQ0004DEX*CEQ0038DEX*FCA0TDPTPM*CDG001BTRM
0.01	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ALHE*CDG001BDGR
0.01	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ADGR*CDG001BLHE
0.01	0.00	0.00	12.97	24.09	25.27	15.56	10.32	7.50	4.26	CEQ0004DEX*CEQ0005DHE*CEQ0038DEX*FCATHRODHE*-CEQ0043DEX
0.01	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ADGS*CDG001BTRM
0.01	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG001ATRM*CDG001BDGS
0.01	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ADGS*CDG001BTRM
0.01	0.00	0.00	10.28	24.13	26.87	16.33	10.48	7.55	4.35	CEQ0004DEX*CEQ0008DEX*CDG001ATRM*CDG001BDGS
0.01	0.00	0.00	8.49	16.93	21.70	16.84	13.99	12.57	9.48	CEQ0001DHE*CEQ0021DEX*FCA0TDPTPM
0.01	0.08	10.03	26.82	24.85	17.95	9.04	5.38	3.74	2.12	CEQ0004DEX*CEQ0007DEX*CDG1ASTCOM
0.01	0.00	0.00	0.00	0.00	59.28	24.83	10.51	4.39	0.99	-CEQ0008DEX*-CEQ0007DEX*CEQ0001DHE*CEQ0034DEX
0.01	0.00	0.00	0.00	0.00	58.12	25.33	10.93	4.59	1.03	-CEQ0008DEX*-CEQ0007DEX*CEQ0001DHE*CEQ0033DEX
0.01	0.00	0.00	0.00	0.00	53.36	27.09	12.68	5.58	1.29	-CEQ0008DEX*-CEQ0007DEX*CEQ0001DHE*CEQ0029DEX
0.01	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPS*CDG001ATRM*CDG001BDGR
0.01	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPS*CDG001ADGR*CDG001BTRM
0.01	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPR*CDG001ATRM*CDG001BDGR
0.01	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCA0TDPTPR*CDG001ADGR*CDG001BTRM
0.01	7.71	41.91	25.68	12.60	6.53	2.71	1.44	0.93	0.50	CEQ0004DEX*FCATHRODHE*CDG001ADGR*CDG001BDGR
0.01	0.00	0.00	33.82	38.67	20.12	5.41	1.50	0.43	0.06	-CEQ0008DEX*-CEQ0007DEX*CEQ0004DEX*CEQ0038DEX*CDG001BLHE*-CEQ0043DEX
100.00	1.21	6.93	9.30	11.79	15.15	13.74	13.71	14.58	13.58	
MEAN = 1.1178E-05										

TABLE 3-5 DOMINANT SEISMIC EVENT SEQUENCES - LLNL CURVE

FREQUENCY CONTRIBUTION TO ACCELERATION INTERVAL										ACCIDENT SEQUENCES
% Contrib.	0.076	0.153	0.255	0.357	0.459	0.561	0.663	0.765	0.918	
7.88	0.00	0.00	2.46	11.70	16.96	17.61	21.60	19.78	9.89	CEQ0004DEX*CEQ0007DEX*CEQ0012DEX
6.84	0.00	0.00	0.55	6.61	14.77	18.52	24.47	23.27	11.81	CEQ0004DEX*CEQ0008DEX*CEQ0012DEX
4.29	1.22	47.52	28.96	14.76	4.83	1.59	0.77	0.30	0.05	-CEQ0008DEX*CEQ0004DEX*CDG001ADGR*CDG001BDGR
3.64	0.00	0.00	6.66	18.64	19.88	16.81	17.67	14.26	6.07	CEQ0005DHE*CEQ0007DEX*CEQ0038DEX*-CEQ0043DEX
3.34	0.00	0.00	0.84	6.20	12.32	16.23	23.96	25.42	15.04	CEQ0004DEX*CEQ0007DEX*CEQ0022DEX
3.08	0.00	0.00	0.18	3.29	10.08	16.03	25.48	28.08	16.86	CEQ0004DEX*CEQ0008DEX*CEQ0022DEX
2.87	0.00	0.00	1.63	11.57	19.02	19.42	21.98	18.43	7.96	CEQ0005DHE*CEQ0008DEX*CEQ0038DEX*-CEQ0043DEX
2.48	0.00	0.00	0.00	3.65	9.59	15.06	24.93	28.57	18.19	CEQ0004DEX*CEQ0007DEX*CEQ0023DEX
2.46	0.00	0.00	26.10	40.40	20.83	7.78	3.56	1.18	0.13	-CEQ0008DEX*-CEQ0007DEX*CEQ00C4DEX*CEQ0012DEX
2.36	0.00	0.00	0.43	4.67	11.29	16.40	25.20	26.79	15.23	CEQ0007DEX*CEQ0019DEX*CEQ0038DEX*-CEQ0043DEX
2.36	0.00	0.00	0.00	1.88	7.61	14.42	25.71	30.61	19.77	CEQ0004DEX*CEQ0008DEX*CEQ0023DEX
2.29	0.00	0.00	0.00	2.30	7.90	14.44	25.89	30.54	18.94	CEQ0007DEX*CEQ0020DEX*-CEQ0043DEX
2.21	0.00	0.00	0.09	2.44	9.10	15.97	26.41	29.17	16.83	CEQ0008DEX*CEQ0019DEX*CEQ0038DEX*-CEQ0043DEX
2.21	0.00	0.00	0.00	1.17	6.19	13.65	26.37	32.30	20.32	CEQ0008DEX*CEQ0020DEX*-CEQ0043DEX
1.98	0.00	0.00	3.09	13.70	18.41	17.83	20.58	17.93	8.47	CEQ0004DEX*CEQ0007DEX*CEQ0038DEX*CDG001BDGR
1.94	0.00	0.00	41.36	37.72	14.31	4.36	1.71	0.50	0.05	-CEQ0008DEX*-CEQ0007DEX*CEQ0005DHE*CEQ0038DEX*-CEQ0043DEX
1.86	0.00	0.00	0.00	0.66	4.86	12.74	26.77	33.71	21.26	CEQ0008DEX*CEQ0013DEX*CEQ0018DEX*-CEQ0043DEX
1.68	0.00	0.00	0.71	7.92	16.41	19.18	23.85	21.58	10.35	CEQ0004DEX*CEQ0008DEX*CEQ0038DEX*CDG001BDGR
1.67	0.00	0.00	0.02	1.75	10.21	20.09	30.62	27.32	9.98	CEQ0008DEX*CEQ0013DEX*CEQ0028DEX*-CEQ0043DEX*-CEQ0037DEX*-CEQ0024DEX
1.65	1.22	47.52	28.96	14.76	4.83	1.59	0.77	0.30	0.05	-CEQ0008DEX*CEQ0004DEX*CDG1ARNCOM

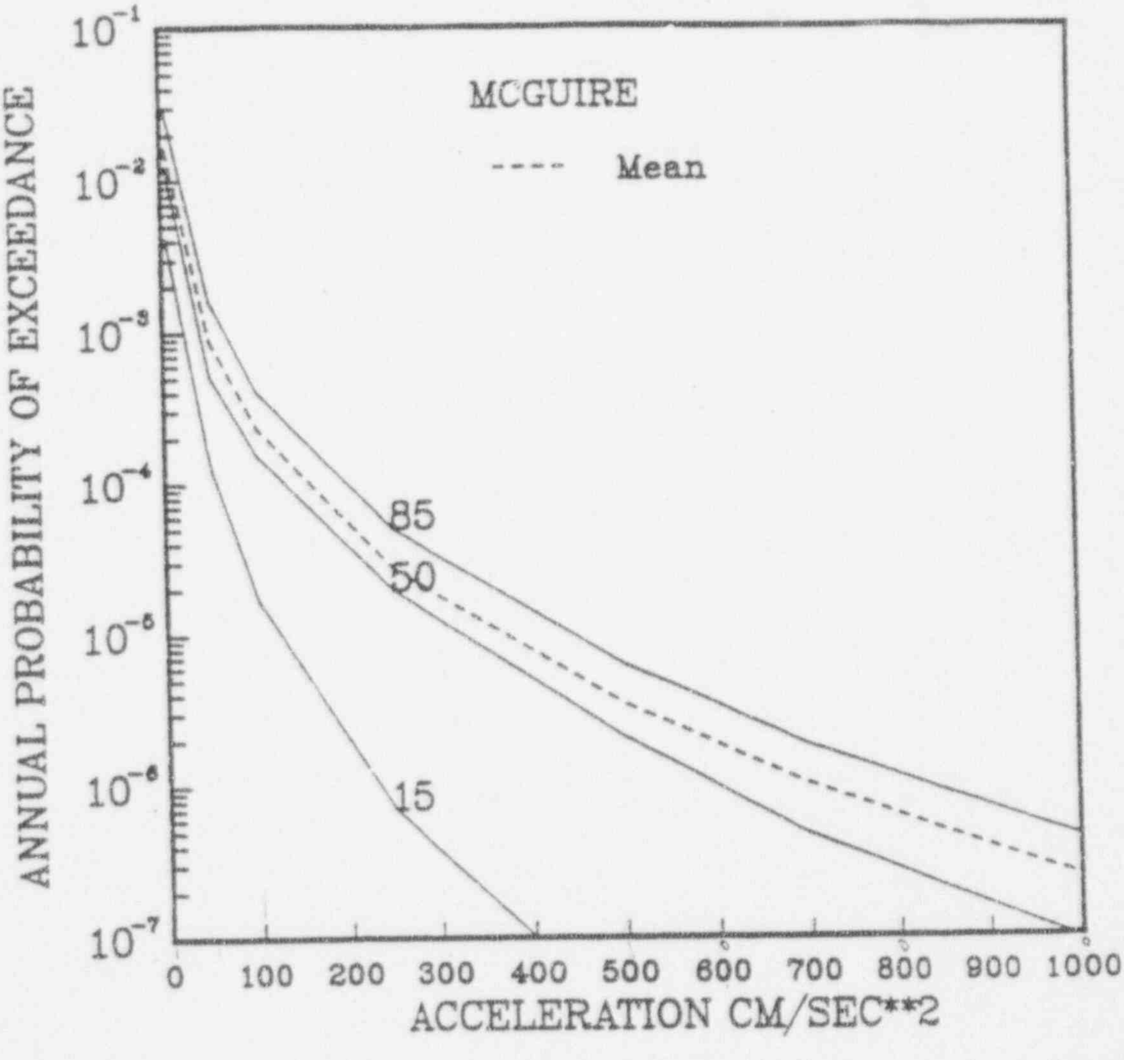

FIGURE 3-1

McGuire Seismic Hazard Curve

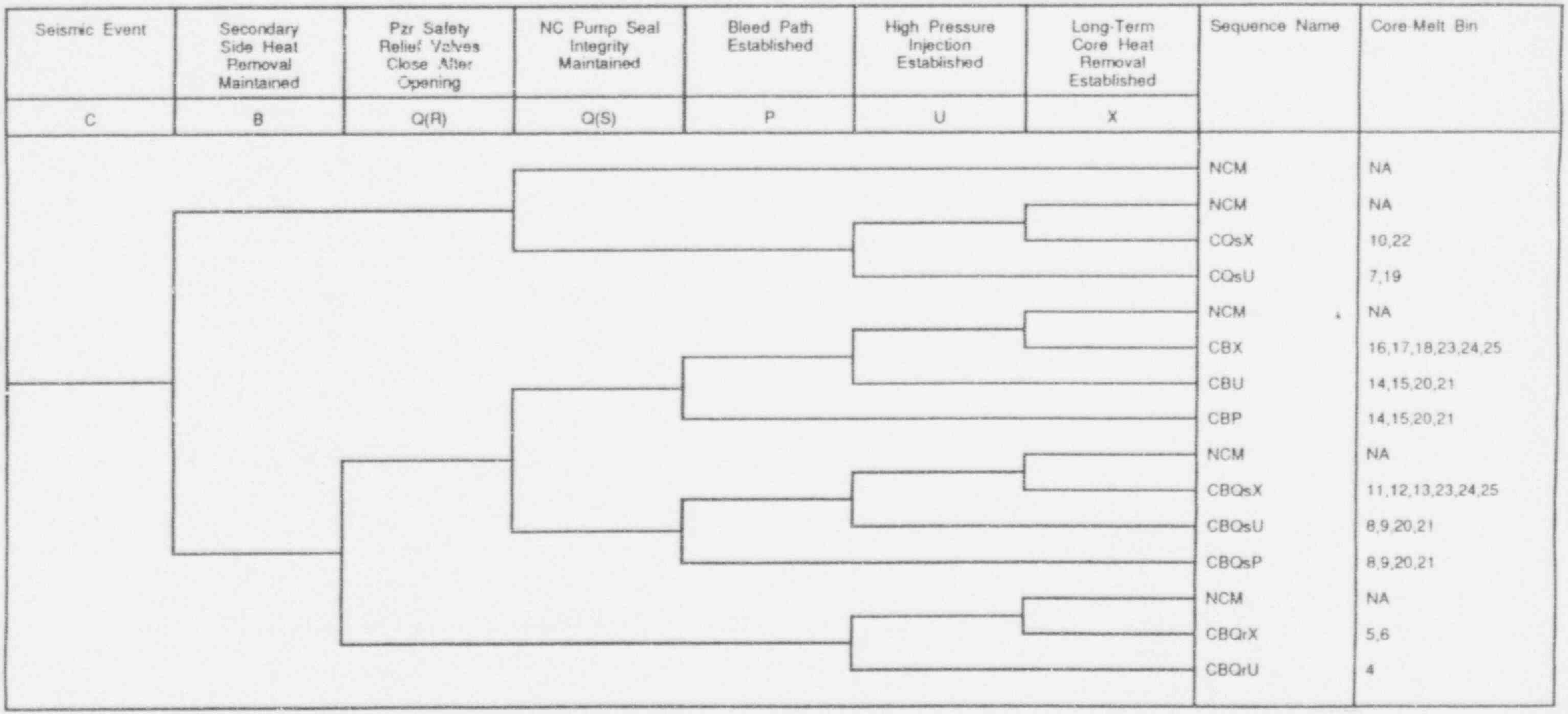

FIGURE 3-2

McGuire Seismic Event Tree

4. INTERNAL FIRE ANALYSIS

4.0 METHODOLOGY SELECTION

The Fire PRA evaluation in section 3.5 of the McGuire Probabilistic Risk Assessment (Ref. 1.2) was used to address Generic Letter 88-20, Supplement 4. An additional evaluation has been done to verify assumptions, technical inputs and conclusions of this portion of the McGuire PRA. A plant walkdown has been used to evaluate issues identified in NUREG CR-5088, "Sandia Fire Risk Scoping Study." These issues include:

- Smoke Generation/Migration Effects
- Water Spray and Migration
- Seismic/Fire Interaction
- Control Systems Interaction, and
- Compartment Interaction Analysis.

The last two issues above were addressed by an evaluation described in sections 4.8.7 and 4.8.8 of this report. The other issues above were addressed in the plant walkdown.

This evaluation and plant walkdown resulted in Revision 2 to the Fire Protection portion (section 3.5) of the PRA. This revision has not identified unacceptable risks. The Sandia Fire Risk Scoping Study Issues have been satisfactorily resolved.

Walkdown checklists are maintained on file (File MCC-1435.03-00-0006).

The plant walkdown is documented in section 4.8 below.

4.1 FIRE HAZARD ANALYSIS

The analysis of fire hazards at McGuire has been done using an event tree approach. This is described in section 3.5 of the McGuire PRA.

4.2 REVIEW OF PLANT INFORMATION AND WALKDOWN

The following plant information has been reviewed in performing the McGuire PRA:

The McGuire Fire Protection Review
The McGuire Fire Protection Safe Shutdown Review
McGuire Fire Area Drawings
McGuire General Arrangement Drawings.

A walkdown of fire areas has been performed for the PRA analysis. The physical arrangement of equipment and the protection features were noted.

The process of identifying critical fire areas is described in section 3.5 of the McGuire PRA. Any area where a fire has the potential to lead to one or more initiating events is defined as a critical fire area and is examined further for erosion of safety margin and spread of fire to other areas.

4.3 FIRE GROWTH AND PROPAGATION

Fire growth and propagation are treated in the McGuire PRA with a multi-stage event tree (Figure 3.5-1 in the McGuire PRA). The parameters for this tree are based on values from NUREG/CR-0654 (Reference 4 in the McGuire PRA) and on fitting the event tree end states to the distribution of growth of fire events estimated from event descriptions in a generic data base (EPRI NP-3179, Reference 2 in the McGuire PRA).

The spread of fire from one area to another is treated by assigning a conditional probability of 1.0E-02 for failure of a fire barrier, given that a fire is large enough to challenge the barrier. This assumption is further described in section 3.5.2.4 of the McGuire PRA.

Expert judgment is used to quantify the parameters for the fire event tree parameters taken from NUREG/CR-0654. The correct set of these event tree parameters is chosen for each fire area based on specific features of the fire area.

Human intervention in detecting a fire or preventing the spread of a fire is accounted for by fire event tree parameters.

Sources of uncertainty in the McGuire PRA fire analysis are described in section 3.5.5 of the McGuire PRA.

4.4 EVALUATION OF COMPONENT FRAGILITIES AND FAILURE MODES

For most components, susceptibility to fire is treated in the McGuire PRA by the fire event tree.

Hot shorts are considered for the control room and cable room. It is estimated (as stated in section 3.5.4 of the McGuire PRA) that cable fires will cause equipment to trip (by energizing a trip coil) before control fuses blow 20% of the time. When control fuses blow, this will fail a breaker in the "as is" condition.

As stated in section 3.5.4 of the PRA, this is a very conservative assumption. Heat from a fire would be expected to be transmitted to conductors by the cable armor which would cause a ground. The ground condition would then cause control fuses to blow.

4.5 FIRE DETECTION AND SUPPRESSION

Fire detection and suppression features are listed for each critical fire area in section 3.5.3 of the McGuire PRA. Credit is taken for features in each area by using different parameters in the McGuire PRA Figure 3.5-1 event tree. The event tree parameters are listed, for each area examined, in Table 3.5-5 in the McGuire PRA.

The initial response to an incipient fire is assumed to occur within 10 minutes of detection. A faster response is assumed for the control room which is continuously occupied. These assumptions were verified as part of the walkdown, described in section 4.8 below, which was used to address Fire Risk Scoping Study Issues.

4.6 ANALYSIS OF PLANT SYSTEMS, SEQUENCES AND PLANT RESPONSE

The plant response for fire scenarios is evaluated with the McGuire Transient Event Tree (Figure 2.2-1 in the McGuire PRA). The plant response to these conditions was evaluated by manipulating the logic for the transient event tree. Non fire induced failures are included in the transient event tree logic.

Fire core damage sequences appear in Table D-7 of Appendix D of the McGuire PRA. This table includes the sequence frequencies and the overall core damage frequency due to fire. The overall core damage frequency from fire initiated events is estimated to be 2.3E-07 per year, based on this IPEEE analysis.

Revised Table D-7 of the McGuire PRA (which is included in Appendix B) lists fire core damage sequences including human reliability events and plant damage states. The top sequence (loss of Nuclear Service Water due to a fire in the Vital I & C area) includes assumptions that offsite power is available and that the Containment Ventilation Cooling Water System will be used to back up Nuclear Service Water. Auxiliary feedwater is assumed to be available for <u>all</u> listed sequences. This assumption implies late core damage. Auxiliary feedwater availability is based on the many redundant control features and suction sources available to the turbine-driven auxiliary feedwater pump.

Other sequences listed in Table D-7 include loss of Control or Cable Room and Main Feedwater Pump fire sequences. For these, offsite power is assumed to be lost.

Human reliability events associated with Table D-7 sequences are described, along with associated assumptions, in section 5 of the McGuire PRA. The only human reliability event listed is a failure to deploy to the SSF in time. The SSF is credited for the Control Room / Cable Room and Vital I & C area fire sequences.

The uncertainty associated with McGuire PRA sequences, including fire sequences, is discussed in section 8.3 of the McGuire PRA.

4.7 ANALYSIS OF CONTAINMENT PERFORMANCE

Containment isolation and containment safeguards assumptions are the same for the fire analysis as for other PRA sequences. The containment safeguards analysis is described in section 4 of the McGuire PRA.

The supplemental walkdown (described in section 4.8 below) has not resulted in the identification of any additional fire related containment failure modes.

4.8 TREATMENT OF FIRE RISK SCOPING STUDY ISSUES

4.8.1 Strategy

As stated above, the IPEEE walkdown used the existing fire PRA and supplemented it with investigation of the Sandia Fire Risk Scoping Study Issues.

4.8.2 Walkdown Team

The walkdown team was comprised of a Fire Protection Engineer from the Duke Power General Office, a PRA Analyst from the Duke Power General Office, a Fire Protection Engineer from McGuire Nuclear Station and the project Program Manager.

The peer review for the walkdown was performed by a Fire Protection Engineer at Catawba Nuclear Station.

4.8.3 PRA Assumptions, Input and Verification

The walkdown included a review of the fire PRA (section 3.5 in the McGuire PRA) to assure that assumptions and inputs were valid and to verify technical conclusions. Validation of frequencies and probability estimates was beyond the scope of the walkdown.

Where the PRA assumptions and inputs were incompatible with station specific arrangements and configurations or conclusions could not be verified, the Fire PRA was revised accordingly. The most significant change to the PRA was the identification of a way to lose Nuclear Service Water from a fire in the Vital I & C area.

This and other modifications resulted in Revision 2 to section 3.5 in the McGuire PRA (see Appendix B). With the changes implemented in the Revision 2 analysis, the walkdown determined that the fire PRA assumptions, inputs and conclusions are accurate and valid. There are no unacceptable risks or outliers identified as a result of the revision to the fire PRA.

4.8.4 Smoke Generation/Migration Effects

Each area was inspected to determine if there is potential for fire to generate an appreciable amount of smoke. Potential fire sources are considered to be: cables with exposed plastic insulation, flammable or combustible liquids in non-seismic containers, in situ or transient (permitted by Station Directive) combustible materials or equipment failure. Where fixed automatic suppression systems are installed, suppression effects are expected to mitigate smoke generation in the area of origin. Smoke is also mitigated by the large volume of certain fire areas.

In areas where smoke may be generated, the walkdown identified adjacent areas (with openings in the boundary) where smoke migration may affect redundant safety related equipment. The possibility of two remote fire areas containing safety related equipment redundant to each other, to which smoke may migrate (from the area under consideration), was also investigated.

Where these adjacent or remote fire areas were identified, the potential for smoke isolation, control and/or exhaust using the installed ventilation system was investigated. In case the ventilation system was unable to control or mitigate unacceptable smoke migration, the walkdown checklist was structured to evaluate fire brigade response and effectiveness in controlling effects of smoke generation and migration.

Smoke effects were evaluated for equipment which may be susceptible to smoke accumulation. This includes electrical devices such as contacts, terminations and relays, and devices such as pressure and temperature transmitters. Smoke does not affect equipment such as enclosed motors, pumps or cables.

The only area where smoke control was identified as a possible concern at McGuire was the Vital I & C battery area on level 733 in the Auxiliary building. It is possible that the fire brigade would approach a fire in the Vital I &C battery area from the Turbine Building through the 4160 V switchgear room. The switchgear room contains SSF cables which would not be affected by smoke.

Station fire brigade training for a fire in the Vital I & C area indicates that fire hoses would be routed through the common area on the 750 level of the Auxiliary Building. The Vital I & C area was already identified as an area where both trains of Nuclear Service water could be affected. For these reasons, smoke migration to the B train switchgear room is judged not to pose significant additional risk.

The walkdown for other fire areas concluded that where appreciable smoke could be generated by fire, existing smoke control capability is sufficient to preclude unacceptable damage.

4.8.5 Water Spray and Migration Effects

The walkdown team investigated the potential risk to safety related equipment due to water discharge from sprinkler heads and fire hoses. The team also evaluated the potential for fire suppression water discharge to migrate from the area of origin and to drip, spray, splash or pond in another area causing damage to redundant safety related equipment.

The walkdown for most fire areas concluded that there are no areas in which the potential for water spray or migration from fire suppression activities created an unacceptable risk to plant safety. An exception to this is the upper 4160 V switch gear room.

The floor between 4160 V switchgear rooms at McGuire is not sealed water tight. This was identified as a concern by the walkdown team. However, by comparing this situation to other modeled PRA events, it was determined that this situation does not present an unacceptable risk to plant safety. This is further discussed in section 3.5.3 of the McGuire PRA (see Appendix B of this report).

NUREG-1472 recommends that Generic Issue 57 (effects of fire protection system actuation on safety-related equipment) be closed out with no new regulatory requirements. It also recommends that cost effective modifications that may be desirable be identified by the IPEEE program.

This issue has now been examined by IPEEE walkdowns of McGuire Nuclear Station. No new cost effective modifications have been identified. Therefore, Generic Issue 57 should be considered closed for McGuire.

4.8.6 Seismic/Fire Interaction

Each area was examined to determine if there is a potential for a seismic event to damage equipment or components resulting in fire ignition, propagation or increased fire hazard. This was accomplished by identifying high energy electrical equipment (i.e., more than 600 V), location of flammable or combustible gas piping and equipment containing more than 5 gallons of combustible or flammable liquids. Where these were identified, existing documentation such as FSAR table 3-7, and requirements of the Duke Power Company Quality Assurance program were used to determine that seismic qualification was adequate to preclude fire seismic interaction during the Safe Shutdown Earthquake. If seismic qualification of this equipment could not be determined, the item would be referred to the seismic margins walkdown team for resolution.

The reactor coolant pump motors were identified as the only pieces of high energy electrical equipment in a fire area of concern which did not have seismic qualification. The potential for fire ignition due to seismic failure of these is described below.

The locations of non-seismic fire protection control panels and actuation devices were reviewed to determine if a seismic event would cause inadvertent system operation. The walkdown determined that automatic, heat activated sprinkler heads are the only devices that may fail and cause actuation. Seismic induced failure of control panels was found not to be a potential problem. Seismic performance of sprinkler heads was evaluated and determined to be acceptable as stated in Reference 4.1.

The potential for fire protection systems water piping failure during a seismic event was investigated as part of the seismic walkdowns discussed in section 3.1.2.3 of this report.

Potential for fire ignition due to seismic failure of reactor coolant pump motors was recognized as a potential risk. A determination was made that fires involving reactor coolant pump motors (inside containment) would not affect ability to achieve safe shutdown because primary system charging and residual heat removal capability would not be affected.

Incore thermocouple cables which are routed in close proximity to some RCP motors are constructed of stainless steel sheathed, mineral insulated cable which is fire resistant.

Any heat release due to reactor coolant pump motor fire would be absorbed and mitigated by ice condensers and the containment spray systems.

4.8.7 Control System Interactions

This concern, identified by Sandia, is primarily focused on plant specific configurations which do not have "independent" remote control or monitoring circuits such that fire in the main control room would disable the remote shutdown capability. (See Reference 4.2 and the McGuire Safe Shutdown Review Manual).

McGuire has a Standby Shutdown System (SSS). The Standby Shutdown Facility is located in the plant yard and is physically independent of the main control room. The Standby Shutdown System uses portions of the Auxiliary Feedwater System. SSS components including the turbine driven auxiliary feedwater pumps and SSS controls and instrumentation are physically protected from fires in other plant areas by fire barriers and electrically protected from malfunction due to fire effects by optical isolators.

Because the SSF is physically and electrically independent of the control room and auxiliary shutdown panel, the control systems interaction issue is considered to have been addressed for McGuire.

4.8.8 Compartment Interaction Analysis

The licensing basis of the McGuire fire protection program is that both units can achieve safe shutdown following fire in any fire area of the plant. There is concern that fire may spread between compartments, by barrier failure, and affect redundant trains of equipment required for safe shutdown.

Since McGuire uses the Standby Shutdown System approach, there are few barriers where such failure could result in fire damage to redundant trains of safe shutdown equipment. The potential risk and frequency of fire barrier failures are considered in the Fire PRA and are determined to be acceptable as stated in References 4.3, 4.4, and 4.5.

4.8.9 Walkdown Conclusions

There are no unacceptable risks or outliers identified by the McGuire IPEEE Fire Protection Walkdown. Details of the investigation of each fire area can be reviewed in calculation number MCC-1435.03-00-0006.

4.9 USI A-45 AND OTHER SAFETY ISSUES

These issues are sufficiently treated in the IPE submittal for McGuire.

4.10 SENSITIVITY STUDIES

Sensitivity studies have been done for the most important events and assumptions associated with the fire PRA.

If the event to fail to use Unit 2 nuclear service water (or remote shutdown) during a fire is increased by a factor of 10 (event FIREFLDREC in Table D-7), it is estimated (assuming Rev. 2 of section 3.5 of the McGuire PRA fire results) that this would increase the total McGuire core damage frequency by about 2.6%. If this event were decreased by a factor of 10, it would decrease the total McGuire core damage frequency by about 0.2%.

The fire core damage frequency for the McGuire PRA includes contributions from the fire in the Vital I & C area, the fire in the Cable Room or Control Room, and the fire in the main feedwater pumps. There is some conservatism in the fire in the Vital I & C area scenario. If this sequence is removed from the fire core damage sequences, this would change the total fire core damage frequency from 2.32E-07 per year to 1.02E-07 per year (see table D-7 in Appendix B of this report).

The supplemental walkdown described in section 4.8 has resulted in the conclusion that some changes to Rev. 1 of Section 3.5 of the McGuire PRA are warranted. These changes have been included in Rev. 2 of Section 3.5 of the McGuire PRA (see Appendix B).

Rev. 2 of the McGuire PRA fire section (3.5) has not identified any unacceptable risks.

4.11 DOCUMENTS

The following documents were used in developing this report and completing the walkdown checklists:

- McGuire Prefire Plans
- Fire Protection Design Basis Document
- Safe Shutdown Review Manual
- General Arrangement Drawings
- Fire Boundary Drawings
- McGuire Probabilistic Risk Assessment (Rev. 1)
- Station Directive - Control of Combustible Material
- Fire Detection (EFA) System Drawings (MC-1762 Series)
- Fire Barrier Penetration Seal Drawings
- Mechanical Flow Diagrams -
    - RF/RY System
    - Flammable/Combustible Gas System
    - VA/ VC Ventilation Systems
- McGuire FSAR Table 3-7, "Electrical Systems and Components Summary of Criteria" (includes Seismic Qualification)
- Emergency Power Distribution Systems Bus Elementary Drawing
- Drawings in series MC-1414 and series MC-1407, RCP Oil and Flammable Gas Piping Drawings

5. HIGH WINDS, FLOODS, AND OTHERS

As mentioned in Section 2.3 of this report, a detailed list of natural and man-made external events were reviewed and screened for applicability to McGuire. A listing of these events is given in Table 5-1 and may be also found in Section 3.1 of the McGuire PRA report. Of these, four were identified that warranted a detailed quantification: earthquakes, fires, tornadoes, and floods.

Transportation and nearby facility accidents were also evaluated in the original PRA report and its revision, but their probabilities of occurrence were found to be very low. Nevertheless, to meet the specifications of NUREG-1407, an evaluation using updated information is also presented for these events.

5.1 HIGH WINDS

The details of the McGuire tornado analysis are presented in Section 3.4 of the PRA report. Three processes were involved in examining tornado hazards. The first was to determine the effects of tornado missiles and high winds on the plant. The second was the development of an event tree which mapped out possible sequences of events following a tornado strike. Finally, sequences were quantified using detailed fault trees to model the event tree.

The TORMIS computer code (Refs. 5.1, 5.2, 5.3) was used to evaluate the effects of tornado missiles on the targets of interest. TORMIS is a Monte Carlo simulation code which can model the response of a plant structural model to a tornado event and calculate the probability of missile strikes on specific targets. The code randomly selects a tornado (including parameters) and a random path orientation and tracks the tornado across the site. Inputs to the model include plant structural design data and possible missiles located on-site.

Plant structural data includes wall thickness, strength, building dimensions, and material of composition. A site walkdown was conducted to determine potential missiles. Examples include cars, trees, signs, and lamp posts. The high voltage (230 kV and 500 kV) transmission lines were not considered susceptible to tornado missiles since they are positioned high above the sources of the missiles.

Category I buildings at McGuire are designed to withstand the wind loadings of a design basis tornado (360 mph) and tornado induced negative pressure differential (3 psi). The probability of experiencing tornadoes of this magnitude at McGuire is considered to be extremely small. Therefore, the effects of high winds on McGuire site buildings were not considered.

On the other hand, transmission lines which bring off-site power from the switchyard are considered to be susceptible to wind loadings. It is assumed for this study that any tornado striking the transmission lines will cause an irreversible loss of off-site power.

The tornado cut sets are presented in Appendix D of the McGuire PRA report. The total core-melt frequency was determined to be 1.93E-05 / yr. All of the tornado-initiated sequences are identical to non-recoverable loss of off-site power sequences. The tornado-induced loss of off-site power is followed by failures of the emergency power system. Emergency power system failures are dominated by failures of the diesel generators to run or start on demand. Diesel unavailability due to maintenance is also a significant contributor.

5.2 FLOODS

The details of the McGuire flooding analysis are presented in Section 3.3 of the PRA report. Flooding from both internal and external sources were reviewed. External flooding occurs from heavy rains or breaches of dams. Internal flooding occurs from breaches of plant water systems located inside plant buildings.

In the McGuire FSAR, it was shown that the station embankment will protect the plant from a combination of the worst case upstream dam failure and half of the predicted maximum precipitation.

The McGuire PRA report refers to a previous analysis of external flooding (Ref. 5.4). This analysis assumes that external flood water from Lake Norman could enter the Turbine Building and flow to the TB basement as it drains. The analysis also assumes that the emergency diesels could be lost (a probability of 0.1) due to the flood. Off-site power is assumed to be lost and no credit is taken for the SSF. The total frequency for the external flood sequence is estimated to be about 5.0E-09 / yr. Because of this low frequency, external floods were deemed not to be a significant contributor to risk. This analysis and the FSAR were used in the current revision to the PRA to dismiss external flooding as a concern.

5.3 TRANSPORTATION AND NEARBY FACILITY ACCIDENTS

Transportation and nearby facility accidents include aircraft crashes; eff ‸cs from military and industrial facilities; water, rail, and highway transportation events; on-site hazardous material inventories; and potential gas pipeline ruptures.

5.3.1 Aircraft Crashes

The Standard Review Plan (SRP), NUREG-800 (Ref. 5.5), Section 3.5.1.6, gives guidance for the evaluation of aircraft hazard potential. Per the SRP, if the following criteria are met, then the probability is considered to be < 1E-07 / yr. by inspection:

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

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

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

The criteria above are evaluated as follows:

a) From Ref. 2.4, Douglas Airport is located approximately 14.5 miles south of the plant; therefore, in order to meet the criterion, the annual number of airport operations must be less than:

$$1000 \times (14.5)^2 = \underline{210{,}250}$$

Table 5-2 shows the results of the 1993 Charlotte Air Traffic Control Tower Traffic Summary (Ref. 5.6). The total annual number of operations for 1993 was <u>531,415</u>, including overflights and landings at secondary airports. (Per a telecon with the Traffic Management Unit (Ref. 5.7), this number is indicative of Charlotte area air traffic activity over the past 3 years.) The above criteria, therefore, is not met.

Because this criteria is not met, we need not evaluate the other two. However, for completeness, we will do so.

b) From Ref. 5.8, there are no military training routes within 5 statute miles of the plant; therefore, this criterion is met.

c) Per Ref. 5.8, an airway is located in close proximity to the station. Ref. 5.10 states the width of a federal airway as 8 nautical miles (or 9.2 statute miles) and hence, from inspection, this airway encompasses the plant. Thus, the criteria is not met.

A detailed analysis of aircraft crash potential must therefore be performed.

Per the SRP, the impact frequency can be determined by the following equation:

$$P_c = C\ N\ A/w$$

where:

P_c = probability of aircraft crash; per yr.
C = inflight crash rate per mile for aircraft using airway
N = total number of yearly flights along the airway
A = target area of site structures sensitive .. aircraft impact; $mi.^2$
w = width of airway (plus twice the distance from the airway edge to the site when the site is outside the airway); mi.

Ref. 1.1 estimates ' . value for C to be 0.233E-09. This number represents a generic accident rate since it takes into account **all** accidents that have occurred in the continental U.S. in which the aircraft was destroyed on impact with ground or water.

Per Ref. 1.1, Section II.4.5.2.2, the original aircraft impact study estimated a value for A, the target area, of $0.0062\ mi.^2$. This number was determined, through the use of a site layout map, by constructing a triangle that enclosed both of the Reactor Buildings and the Auxiliary Building.

From Ref. 5.7, air traffic into and out of the Douglas Airport control area (Class B airspace) primarily travels along four corridors extending southeast. southwest. northeast, and northwest. Some traffic would also travel along the normal airways extending north, south, east, and west. Traffic may be assumed to be evenly distributed among each corridor. Thus, with the total annual number of operations for 1993 being 531,415, including overflights and landings at secondary airports, each corridor's annual number of flights is 531,415 / 4 = 132,854.

From Ref. 5.8, air traffic in the vicinity of Catawba would use airways V37 north, and V454 northeast. Consequently, assuming 90% of these flights are conducted along the northeast corridor, the corresponding number of flights along these airways is:

V37 north = (0.10) (132,854) = 13,285
V454 northeast = (0.90) (132,854) = 119,569

Per Ref. 5.8, the centerline of V37 passes approximately 1.2 miles (1.0 nautical mile) from the plant and V454 passes approximately 8.7 miles (7.6 nautical miles) from the plant. Again, Ref. 5.10 states the width of a federal airway as 8 nautical miles. Thus, V37 encompasses the station and w = 9.2 statute mi. The edge of the V454 airway, however, does not encompass the plant. From the above definition, the airway width thus becomes:

w = 9.2 mi. + 2 (8.7 mi. - 9.2 mi. / 2) = 15.3 mi.

Therefore, the total crash probability will be the summation of probabilities for each airway. Returning to our equation,

$$P_c = C\ N\ A/w$$

$$= C\ [\ N_{V37}\ (A/w)_{V37} + N_{V454}\ (A/w)_{V454}\ ]$$

$$= (0.233\text{E-}09)\ \times \left[13{,}285\left(\frac{0.0062}{9.2}\right)+119{,}569\left(\frac{0.0062}{15.3}\right)\right]$$

$$= \underline{1.3\text{E-}08\ /\ \text{yr.}}$$

This impact frequency does not represent the frequency of core melt due to aircraft impact. It merely shows the probability of a plane crash in the target area of McGuire. In order to derive the core-melt frequencies, a complex analysis of consequences would be necessary. Because the frequency of a crash is very small, the probability of a crash *and* a core melt is negligible and, therefore, the event is not considered further for core damage assessment.

5.3.2 Transportation Events

The major north-south transportation corridors in the vicinity of the site are U.S. 321, located approximately 15 miles west of the site; N.C. 16, located approximately three miles west of the site; and I-77, located approximately five miles east of the site. The major east-west transportation corridors are I-40, located approximately 25 miles north of the site; and I-85, located approximately 12 miles south of the site. N.C. 150, located approximately 11 miles northwest of the site; and N.C. 73, located approximately 0.4 miles south of the site, are primarily used by local residents, commuters, and for recreational access to Lake Norman.

Gasoline and oil tank trucks, for local delivery to service stations, marinas, and homes; as well as other types of tanker trucks; can, at the driver's discretion, be expected to travel via N.C. 73 or I-77. The shipment of hazardous materials is regulated by the US Dept. of Transportation with regard to materials loading and unloading (49CFR 173.30 and 49CFR 177, Part B); container requirements (49CFR 173.24 and 49CFR 477.812); quantity limitations (49CFR 173.26); the reporting of accidents (49CFR 177, Part D); and the specification of containers (49CFR 178).

Thus, based upon the regulations noted above and the proximity of alternate major high speed highways bypassing the site, the probability of McGuire being affected by shipment of hazardous materials on highway N.C. 73 is deemed as insignificant.

After reviewing the above screening information as presented in the McGuire FSAR and PRA report, it is determined that no substantial changes to this data have occurred which would affect the results. Thus, the original screening of this event remains valid.

5.3.3 Impact of Nearby Military and Industrial Facilities

Per Ref. 5.9, Appendix A, there are no military or industrial facilities within a 5 mile radius of the plant. Thus, the original screening of this event, as presented in the PRA, remains valid.

5.3.4 On-Site Storage of Toxic Materials

Per Ref. 5.9, App. A, there are no industries within 5 miles of McGuire which transport or store products harmful to the station. On the other hand, small quantities of miscellaneous chemicals capable of producing toxic gases are stored on-site for various plant operations. Per telecons with knowledgeable site personnel (Ref. 5.11 and 5.12), the information below was obtained.

Gaseous chlorine is used for drinking water purification and sanitary waste treatment at the McGuire site. No individual container on the site will contain more than 150 pounds of chlorine. There will normally be six 150 pound containers in use, with additional containers on hand at any given time. Normally, these containers will number about 20 or less. (This information verifies the existing FSAR material.)

Aqueous ammonia is used to increase steam generator water chemistry pH in preparation for wet layup. Typically, two 55 gallon drums per unit are stored in the Turbine Building.

Hydrazine is used in various applications as an oxygen scavenging agent. There are typically no more than four 55 gallon drums per unit stored in the Turbine Building at any one time.

Ethanolamine (ETA) is a pH control additive. No more than four 55 gallon drums are stored in the Turbine Building.

It is unlikely that leaks from these containers would result in dangerous concentrations in the Control Room; however, the Control Room can be isolated from the outside environment, if necessary, and portable breathing equipment is available. Based on this information, it is concluded that potential hazards from the storage of toxic materials on-site is minimal.

5.3.5 On-Site Storage of Explosive Materials

The greatest potential for hazards of this nature is the storage of hydrogen, oxygen, and nitrogen.

An analysis of hydrogen tank storage has been performed via Duke study ISA 83-09 (Ref. 5.14). This evaluation reviewed the consequences of tank failure and the subsequent damage to plant walls. It was postulated that a hydrogen burn / explosion could propel a tank towards the Turbine Building, Doghouse, Auxiliary Building, or the Diesel Building. Since there is no safety-related equipment in the Turbine Building, a tank impact will not affect the plant's ability to shutdown. Similarly, the plant can withstand a main steam line break should the tank hit and enter the Doghouse. The Auxiliary Building can also withstand missile penetration without compromising plant shutdown. Finally, the Diesel Building missile barrier was analyzed and was found to able to sustain an impact without damage to the safety-related equipment inside.

Furthermore, McGuire Design Study MGDS-0229/00 (Ref. 5.15) reviewed the potential for hydrogen leaks and accumulation in the Auxiliary Building per NRC Information Notice 87-20. The study showed that the hydrogen concentration level in the affected areas is several orders of magnitude lower than the 4% criteria established by the NRC. Thus based upon these reviews, it is concluded that a core-melt due to hydrogen storage or usage is virtually non-existent.

Per discussions with McGuire Fire Protection (Ref. 5.13), oxygen and nitrogen do not pose explosive hazards.

Based on this information, it is concluded that potential hazards from the storage of these materials on-site is extremely remote.

5.3.6 Gas Pipeline Ruptures

Current gas pipeline maps of the area around the McGuire plant site were reviewed and indicated that no changes to the original PRA screening information as contained in the FSAR had occurred; thus, the original screening of this event remains valid.

5.4 OTHERS

The information presented in Section 3.1 of the McGuire PRA for the remaining external events was reviewed and the information presented therein was determined to remain applicable. Table 5-3 lists each event and the reasoning for screening out that event.

TABLE 5-1

Preliminary External Initiating Event List

1. Aircraft
2. Avalanche
3. Coastal Erosion
4. Drought
5. External Flooding
6. Extreme Winds and Tornadoes
7. Fire
8. Fog
9. Forest Fire
10. Frost
11. Hail
12. High Tide, High Lake Level, or High River Stage
13. High Summer Temperature
14. Hurricane
15. Ice Cover
16. Industrial or Military Facility Accident
17. Internal Flooding
18. Landslide
19. Lightning
20. Low Lake or River Water Level
21. Low Winter Temperature
22. Meteorite
23. Pipeline Accident (gas, etc.)
24. Intense Precipitation
25. Release of Chemicals in On-site Storage
26. River Diversion
27. Sandstorm
28. Seiche
29. Seismic Activity
30. Snow
31. Soil Shrink-Well Consolidation
32. Storm Surge
33. Transportation Accidents
34. Tsunami
35. Toxic Gases
36. Turbine-Generated Missile
37. Volcanic Activity
38. Waves

TABLE 5-2

1993 Charlotte Air Traffic Summary

	Primary Airport		Secondary Airport		Overflights	
	Instrument Operations	TCA Operations	Instrument Operations	TCA Operations	Instrument Operations	TCA Operations
Air Carrier	246,468	0	0	0	3434	0
Air Taxi	130,478	3772	141	0	4204	2147
General	46,183	19,977	17,298	0	22,726	27,890
Military	4203	569	79	0	1270	576
TOTALS	427,332	34,318	17,518	0	31,634	30,613

GRAND TOTAL - 531,415

TABLE 5-3

Screening Justification for Other External Initiating Events

	Event	Remarks
1	Avalanche	There are no mountains in the vicinity of McGuire from which a significant avalanche could be generated.
2	Coastal Erosion	McGuire is located more than 150 miles from the nearest coastal area. However, to protect the lake edge from erosion, the yard areas subjected to waves are protected by riprap underlaid by a thick subgrade of filter material. Therefore, lake edge erosion will not be a significant problem.
3	Drought, High Summer Temps., Low Lake or River Water Level	The effect of a drought, high summer temperatures, low lake level, or low river water level at McGuire is insignificant because there are upstream dams that provide water level control on Lake Norman.
4	Fog	Accident data involving surface vehicles or aircraft would include the effects of fog.
5	Forest Fire	Bush and local forest fires are handled by the local fire department. Such fires are not considered to have any impact on the station because the site is cleared and the fire can not propagate to station buildings or equipment.
6	Frost, Hail, Snow, Ice Cover	Both the Reactor Building and the Auxiliary Building are designed for a combination of snow, ice, and rain. Low winter temperatures causing failure of instruments is included in the plant trip frequency data.
7	Hurricane	The effect of water from a hurricane is considered similar to the effect of intense precipitation.
8	Landslide	Landslides are considered an insignificant hazard at McGuire. The Standby Nuclear Service Water Pond (SNSWP) dam is the only natural or man-made slope which, upon failure, would prevent safe shutdown of the plant. Therefore, the SNSWP was statically designed for stability under all loading conditions..

TABLE 5-3 (cont.)

Screening Justification for Other External Initiating Events

	Event	Remarks
9	Lightning	The most probable effect of lightning is the loss of off-site power due to a strike in the switchyard. These occurrences are accounted for in the loss of off-site power initiating event frequency.
10	Meteorite	This event has a significantly lower frequency than other events with similar uncertainties. The occurrence of a meteorite event could not result in worse consequences than other external events of a higher frequency. Therefore, this event is excluded because it will not significantly influence the total risk.
11	Intense Precipitation	Per the FSAR, the station embankment will protect the plant from a combination of the worst case upstream dam failure plus half of the predicted maximum precipitation. Thus, external flooding from intense precipitation effects alone are enveloped.
12	River Diversion	No present means exist to divert or reroute the river flow through the dams other than insignificant amounts of water used for municipal supply.
13	Sandstorm	McGuire is located more than 150 miles from the nearest area with a large sand deposit. The likelihood of occurrence is insignificant.
14	Seische	Since the flood examined in the FSAR uses the largest rate and volume (for external sources), this analysis provides a reasonable estimate of the effects of all TB flooding events.
15	Soil Shrink-Well Consolidation	Per Ref. 2.4, hazards associated with soil shrink-well consolidation will be insignificant.

TABLE 5-3 (cont.)

Screening Justification for Other External Initiating Events

	Event	Remarks
16	Storm Surge	Since the flood examined in the FSAR uses the largest rate and volume (for external sources), this analysis provides a reasonable estimate of the effects of all TB flooding events.
17	Tsunami	McGuire is located more than 150 miles from the nearest coastal area at an elevation of 760 ft. mean sea level. Therefore, tsunami effects are insignificant.
18	Turbine-Generated Missile	The majority of the structures at McGuire are located either along or within close proximity to the longitudinal centerlines of the respective turbines. Calculations on turbine missiles prepared for the McGuire FSAR indicate that the contribution to plant risk from the turbines would be insignificant.
19	Volcanic Activity	No active volcanoes exist within the vicinity of McGuire.
20	Waves	Since the flood examined in the FSAR uses the largest rate and volume (for external sources), this analysis provides a reasonable estimate of the effects of all TB flooding events.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE PROGRAM ORGANIZATION

The IPEEE program has been managed and performed by Duke Power personnel. The team consisted of engineers from the Severe Accident Analysis, Civil Engineering and Fire Protection groups in the Nuclear Services Section and engineering and operations personnel from McGuire Nuclear Station. The team included individuals with expertise and experience in PRA methodology, seismic capability evaluations, fire protection, and systems engineering. The IPEEE program resulted in updates to the existing PRA that already included external events.

The external events analysis was originally performed as part of the original probabilistic risk assessment of McGuire Nuclear Station that was completed in 1984. The study was performed by Duke Power Company staff with Technology for Energy Corporation as a contractor. Law Engineering Testing Company and Structural Mechanics Associates provided specific input to the seismic analysis. A large-scale review and update of the original study was submitted in 1991. The original study and subsequent update form the basis for the IPEEE.

Duke Power Company's initial staffing to enable large scale PRA and reliability studies in-house began in 1980. A severe accident analysis group was organized and charged with the responsibility to plan, conduct, and coordinate all proposed PRA studies and to maintain and update the plant PRA models as appropriate. In addition to PRA studies, this group is also utilized for engineering support involving severe accident input in such areas as emergency planning, plant design changes and plant operational problems.

6.2 COMPOSITION OF INDEPENDENT REVIEW TEAM

In addition to the independent reviews performed during the original PRA, the work of the IPEEE team has been reviewed by an independent group consisting of senior level engineers and managers within the company. These reviewers have combined extensive experience in seismic design and qualification, fire protection, systems engineering and PRA methodology. In addition to their experience, four members of the peer review team have attended the EPRI sponsored "Walkdown Screening and Seismic Evaluation Training Course" and three have attended the "Seismic IPE Training Course." A list of the peer review team members is included in Table 6-1.

6.3 AREAS OF REVIEW AND MAJOR COMMENTS

The Peer Review Team (PRT) has reviewed aspects of the work performed by the IPEEE team to validate both the process and its results and to ensure proper documentation of the work. This effort included 1) review of the process for selecting areas and equipment for evaluation as well as a review of specific lists, 2) review of process used to evaluate areas identified for review, 3) participation on sample walkdowns to review judgments made in the field, 4) review of the backup documentation and calculations and 5) review and endorsement of the final results and report.

Table 6-2 lists major comments from these reviews.

6.4 RESOLUTION OF COMMENTS

Each comment raised by the PRT was responded to by the IPEEE team and resolved to the satisfaction of the PRT.

TABLE 6-1

PEER REVIEW TEAM MEMBERS

SEISMIC

P. M. Abraham	Section Manager	Severe Accident Analysis	DPCo.
R. L. McCoy	Senior Engineer	Oconee Civil Engineering	DPCo.
J. M. Richards	Senior Engineer	Civil Engineering	DPCo.
W. B. Shoemaker	Senior Engineer	Catawba Civil Engineering	DPCo.
J. E. Thomas	Engineering Manager	Catawba Electrical Engineering	DPCo.

FIRE

S. R. Christopher	Senior Engineer	Catawba Civil Engineering	DPCo.

Table 6-2

Peer Review Team Comments and Resolutions

Comment	Resolution
Document concurrence of Operations personnel regarding system and equipment selection.	Formal concurrence was obtained from operating personnel involved in review.
Add a designation on the Safe Shutdown List for non-QA Condition Equipment and scrutinize such equipment more carefully.	SSELs now indicate which equipment is non-safety. All SSELs will be kept as part of the utility's internal documentation.
Look at valve drawing information to review operator weight/length screening criteria for valves not actually observed in the plant.	Drawing information concerning operator length/weight was reviewed for all unit 1 valves which were not walked down. No problems were found. Since no Unit 1 problems were found, the unit 2 valves which were not walked down were not given the same drawing review.
Document basis of judgments of the Seismic Review Team as much as possible.	Basis of judgments documented as appropriate.

7. PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

The results of the accident sequence analysis is examined to identify those sequences which significantly contribute to the core melt frequency. For reporting purposes, accident sequences are combined into functional sequences. Those with frequencies greater than or equal to 1E-06 / yr. are identified with their initiating events and contributing system failures.

The IPE results were further examined with the objective of identifying potential plant enhancements and to further consider those enhancements which showed merit with respect to core melt frequency. The process of identifying, evaluating, and implementing the various plant enhancements is discussed in Section 3.0 of the IPE submittal.

No fundamental plant weaknesses or vulnerabilities with regard to external events were identified during this examination. Several enhancements to the plant were recommended as a result of the walkdowns. Some are currently being implemented, and others are being considered. They are listed below:

- Add spacers between Unit 1 Diesel Generator batteries and racks.
- Add grout between Component Cooling Heat Exchangers saddle base and concrete curb.
- Trim grating around Steam Vent valves.
- Replace missing bolts on Unit 2 Upper Surge Tanks.
- Modify Unit 2 Turbine Driven Auxiliary Feedwater Pump Control Panel to avoid seismic interaction with pipe.
- Replace or clean and recoat corroded nuts on Auxiliary Feedwater Condensate Storage Tank anchor bolts.
- Tighten arc barrier connections inside Main Control Boards.

In addition to the hardware enhancements, procedural guidelines are being developed to secure movable equipment and structures to prevent potential seismic interactions.

8. SUMMARY

This report details the methodology, implementation, and results of the supplemental examination of external events for severe accident vulnerabilities at McGuire Nuclear Station. This work has been completed by using the existing McGuire PRA, which already included external events, updating it as appropriate and performing the additional enhancements recommended in NUREG-1407.

The major finding from this examination is that there are no vulnerabilities to severe accident risk from external events. Tornadoes and seismic events are the most significant external event contributors to core-melt risk. For both hazards, the primary accident sequences involve a loss of off-site power with diesel generator failures, thereby resulting in a loss of all ac power. There were no plant changes identified that would significantly reduce the risk from external events. Enhancements to the plant were identified during the review and they are currently being implemented.

The IPEEE effort was completed using in-house expertise, resulting in maximum benefit to the company staff in (1) developing an appreciation of severe accident behavior, (2) understanding the most likely severe accident sequences, (3) gaining a qualitative understanding of the overall likelihood of core damage and radioactive material release.

Several generic issues and unresolved safety issues were addressed and considered closed out as a result of the previous PRA work and the IPEEE effort, including:

- USI A-45, "Shutdown Decay Heat Removal Requirements"
- GI 131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants"
- Eastern U.S. Seismicity Issue
- USI A-17, "System Interactions in Nuclear Power Plants"
- NUREG/CR-5088, "Fire Risk Scoping Study"
- GI 57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"
- GI 103, "Design for Probable Maximum Precipitation (PMP)"

Thus, the examination for external event severe accident vulnerabilities, as requested by the NRC via supplement 4 of Generic Letter 88-20, has been completed for McGuire Nuclear Station, Units 1 & 2. The objectives of this program have also been satisfied for McGuire.

9. REFERENCES

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1.2 Duke Power Company, "McGuire Nuclear Station Unit 1 Probabilistic Risk Assessment," November 1991.

1.3 USNRC, Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50-54(f)," November 23, 1988.

1.4 Duke Power Company, "McGuire Nuclear Station IPE Submittal Report," November 1991.

1.5 H. B. Tucker, Letter to USNRC, "Response to Generic Letter 88-20, Supplement 4," Duke Power Company, December 18, 1991.

1.6 T. A. Reed, Letter to T. C. McMeekin, "Review of Response to Generic Letter 88-20, Supplement 4," USNRC, June 16, 1992.

1.7 USNRC, Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10 CFR 50-54(f)," June 28, 1991.

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

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Section 2.0

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2.5 EPRI, " RP101-53, Probabilistic Seismic Hazard Assessment for McGuire Nuclear Station." Palo Alto, California, April 1989.

Section 3.0

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3.2 EPRI, NP-6041, Rev. 1, "A Methodology of Assessment of Nuclear Power Plant Seismic Margin," August 1991.

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3.5 Structural Mechanics Associates, SMA 19201.01, "Seismic Fragilities of Structures and Components at the McGuire Nuclear Station," September 1983.

3.6 Veneziano, D., "Seismic Fragility of the SNSW Pond Dam at McGuire Nuclear Station," Massachusetts Institute of Technology, March 1983.

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3.8 Documentation of the Seismic Event Impact Sequence Model (SEISM) Computer Code, PSA-84-17, Duke Power Company, September 1984.

3.9 Commonwealth Edison, "Zion Probabilistic Safety Study," 1981.

3.10 USNRC, NUREG/CR-3263, "A Comparison of Methods for Uncertainty Analysis of Nuclear Power Plant Safety System Fault Tree Models," April 1983.

Section 4.0

4.1 Memo To File, dated 3/1/94, by John M. Richards, Subject: McGuire Nuclear Station, Seismic Evaluation of Sprinkler Heads for IPEEE, File Nos: MC-1435.00, NSD-0183.

4.2 Correspondence To Conrad E. McCraken, USNRC, from Raymond N. Ng, NUMARC, dated August 4, 1989.

4.3 Memo To File, dated February 7, 1994, by Leo Kachnik, Subject: SSF Disconnect Enclosures, File: MC-1535.00.

4.4 Memo To File, dated January 14, 1992, by Leo Kachnik, Subject: McGuire and Catawba PRA Fire Analysis, Files: CN-1535.00, MC-1535.00.

4.5 Memo To File, dated March 5, 1992, by Leo Kachnik, Subject: Compartment Interaction Analysis, Files: MC-1535.00, CN-1535.00, OS-203.

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5.6 Charlotte Air Traffic Control Tower (Traffic Management Unit); 1993 Traffic Summary

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5.8 Charlotte Sectional Aerial Nautical Chart, 55th Edition, 2/3/94.

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5.10 1988 Federal Aviation Regulations for Pilots, Part 71.

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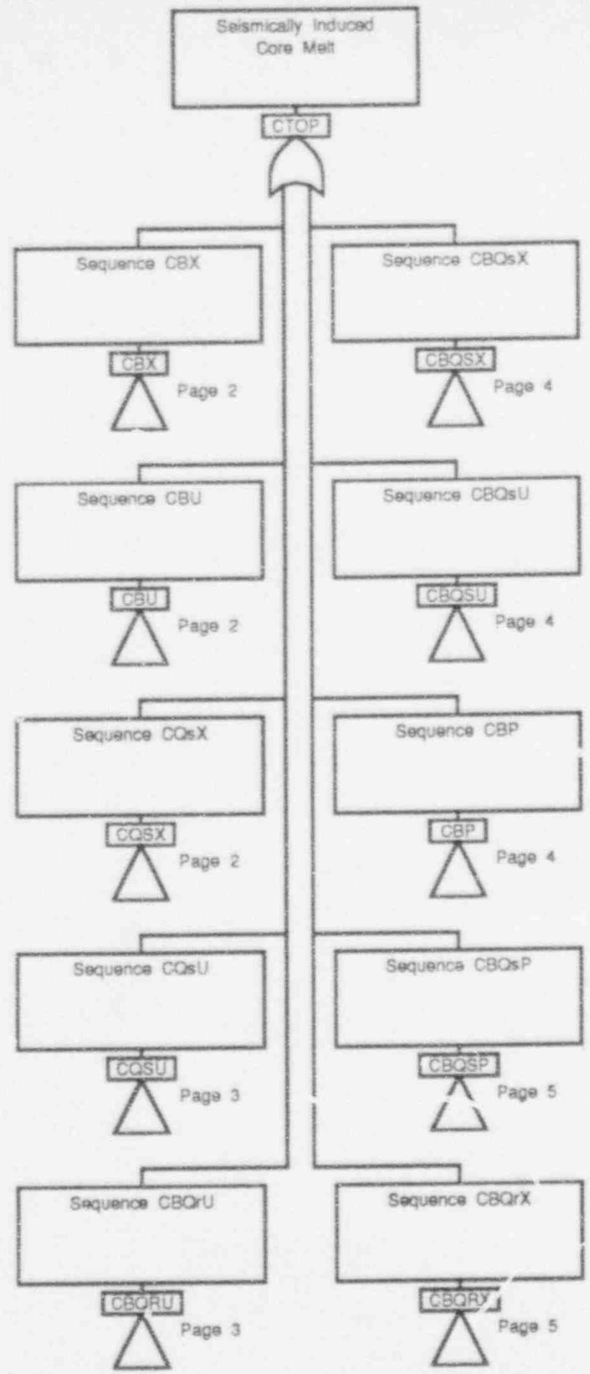
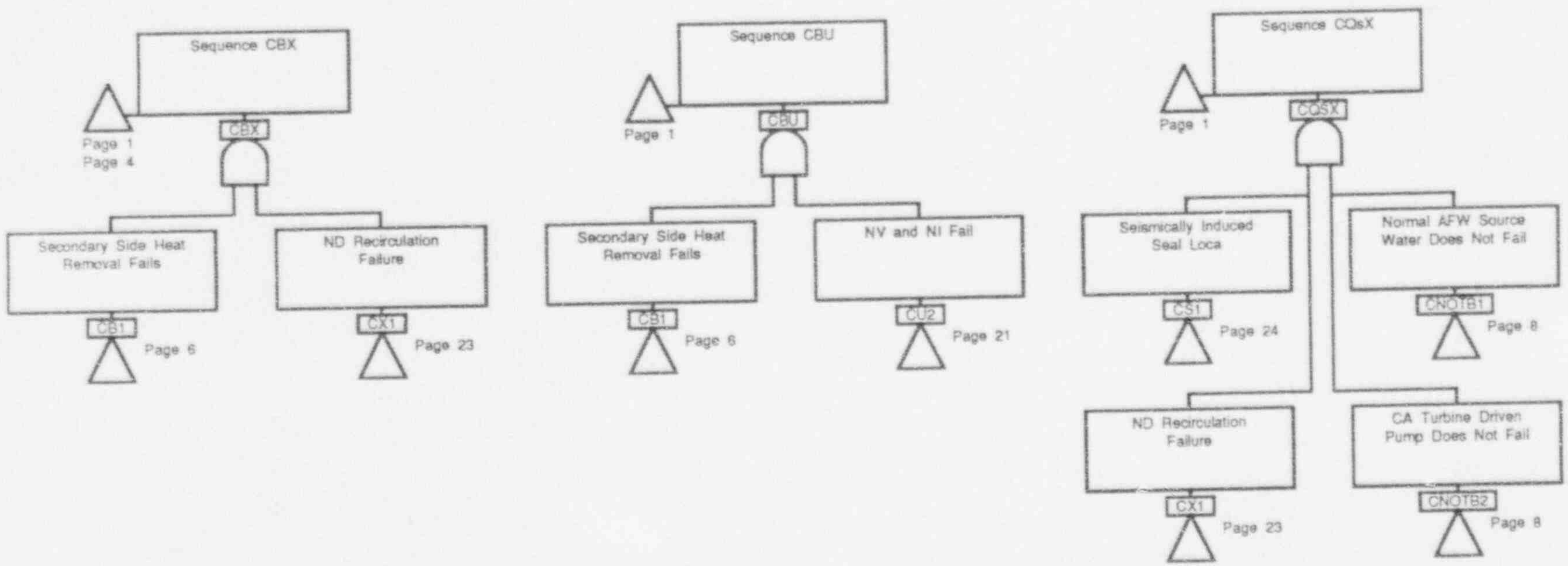
APPENDIX A

MCGUIRE SEISMIC FAULT TREE



McGuire Seismic Fault Tree	.\TREE\MSEISREV.CAF	5-31-94	Page 1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CB1
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CX1
Page 23
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CBU
Secondary Side Heat Removal Fails
CB1
Page 6
NV and NI Fail
CU2
Page 21
Sequence CQsX
Page 1
CQSX
Seismically Induced Seal Loca
CS1
Page 24
Normal AFW Source Water Does Not Fail
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Page 8
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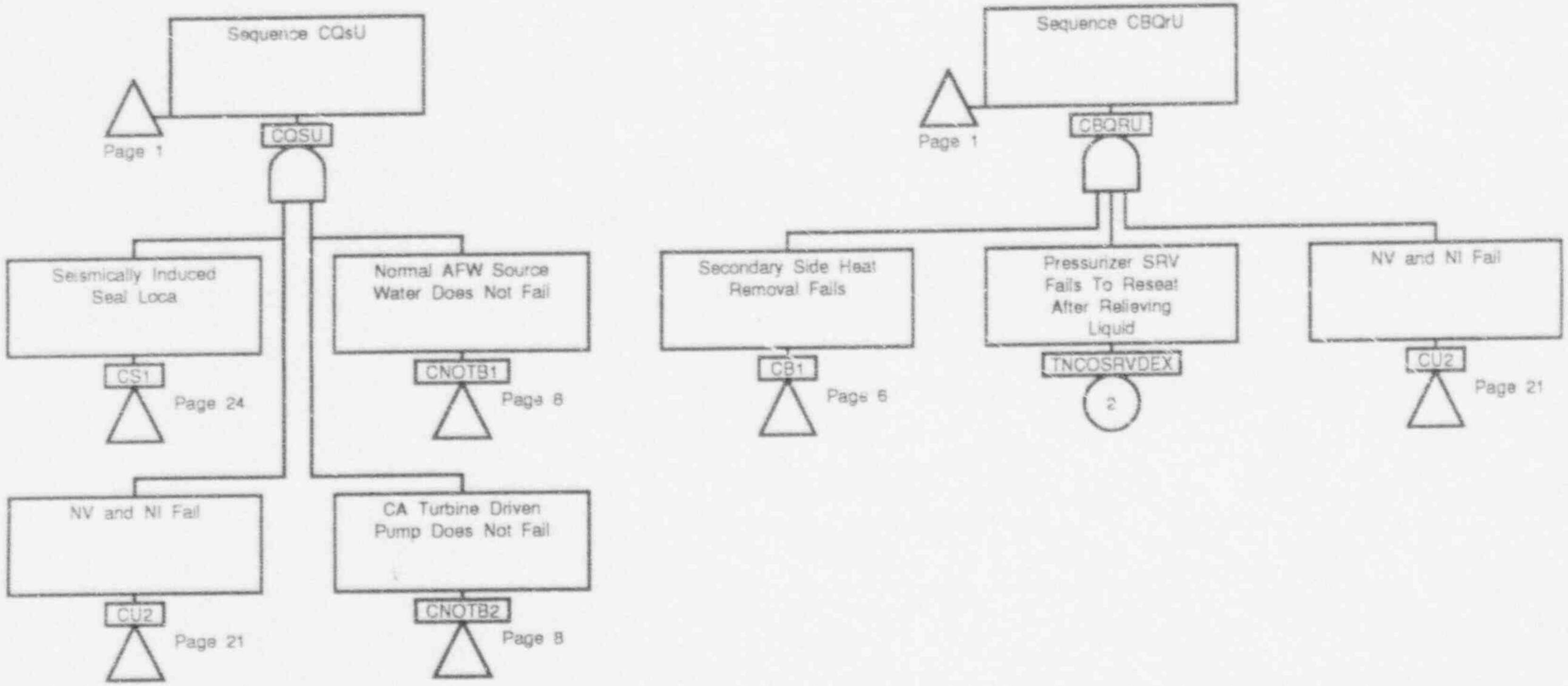
CX1
Page 23
CA Turbine Driven Pump Does Not Fail
CNOTB2
Page 8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CU2
Page 21
CA Turbine Driven Pump Does Not Fail
CNOTB2
Page 8
Sequence CBQrU
Page 1
CBQRU
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CB1
Page 6
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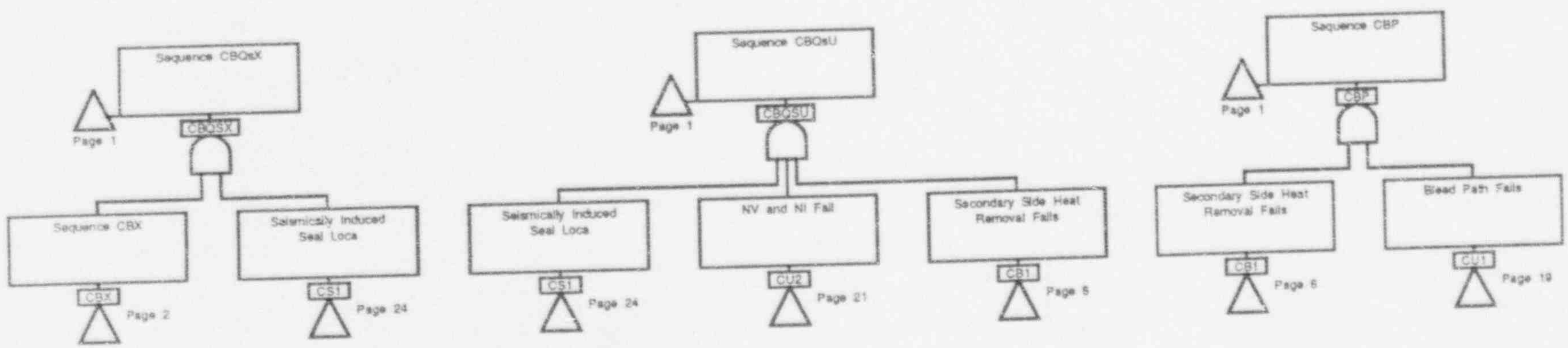
CU2
Page 21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Seismically Induced
Seal Loca
CS1
Page 24
NV and NI Fail
CU2
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Secondary Side Heat
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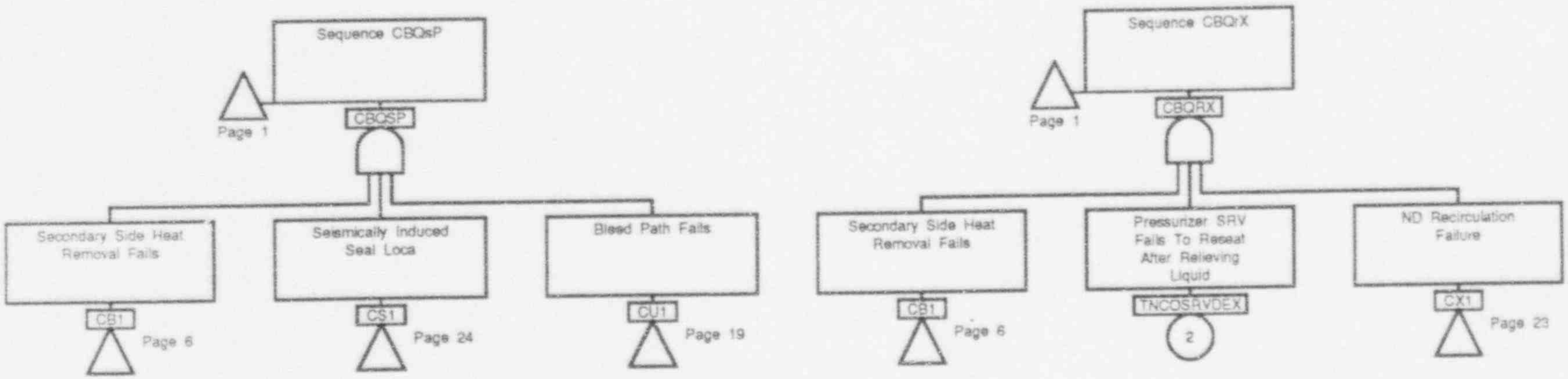
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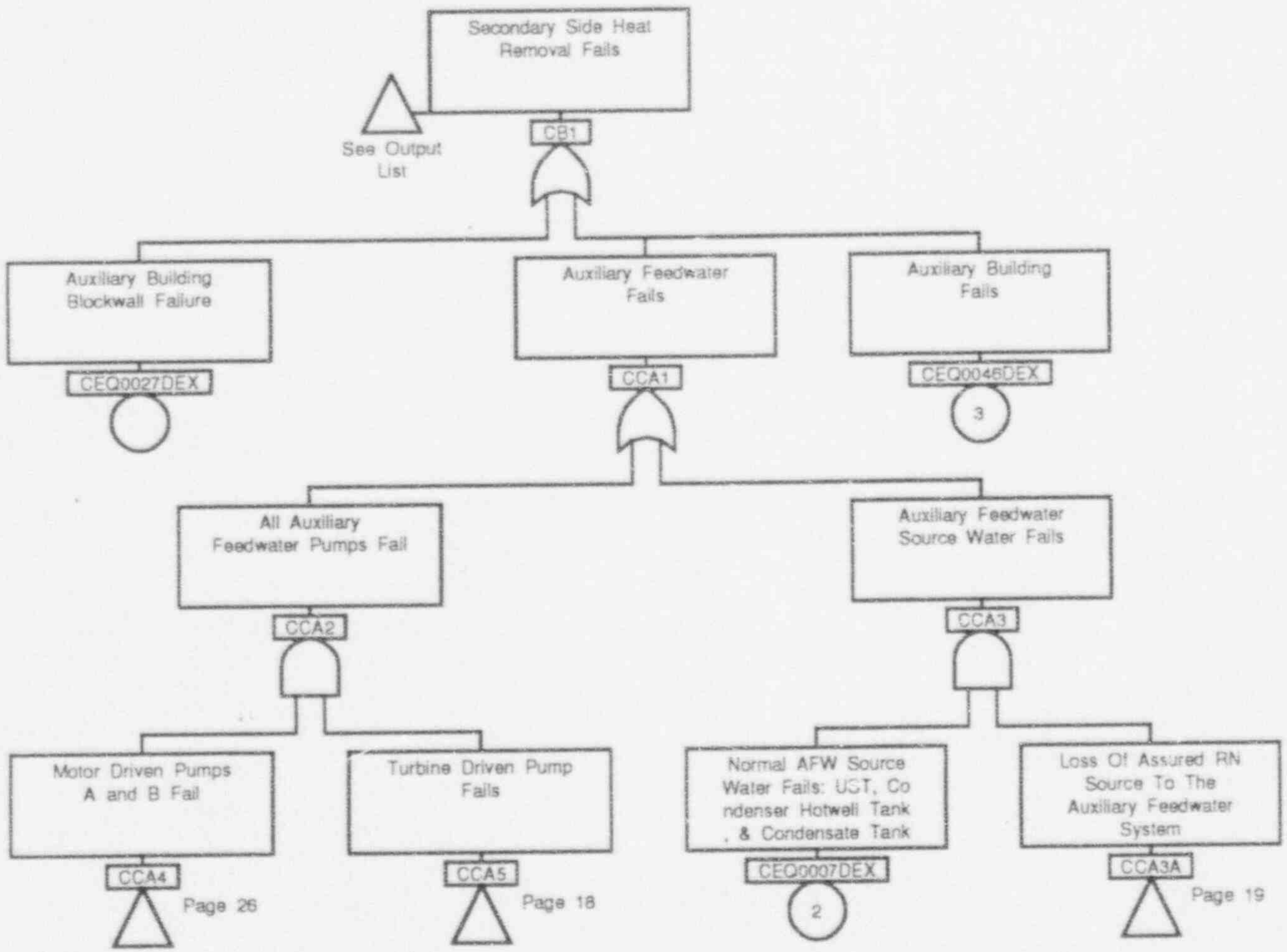
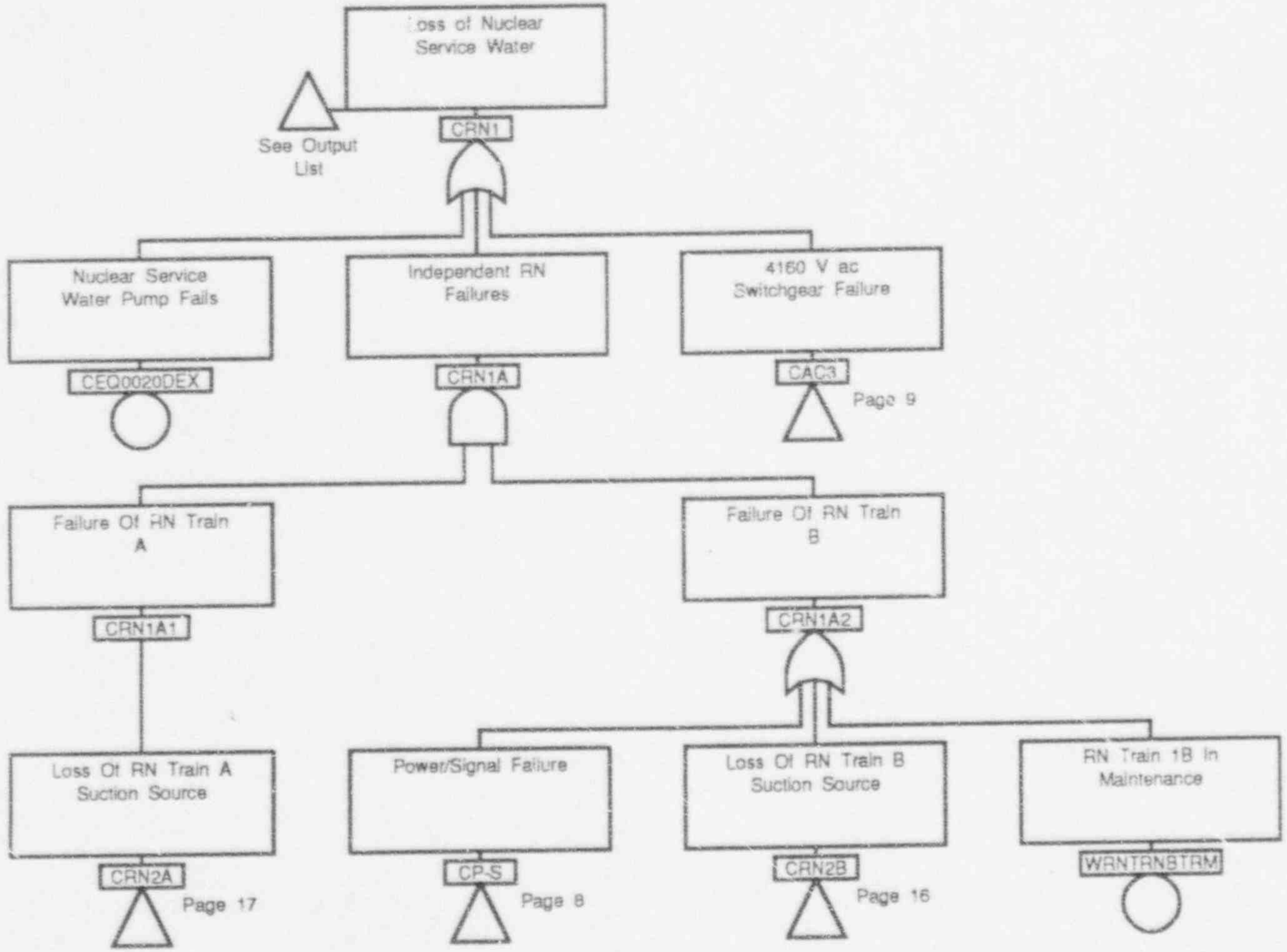
CS1
Page 24
Bleed Path Fails
CU1
Page 19
Sequence CBQrX
Page 1
CBQRX
Secondary Side Heat Removal Fails
CB1
Page 6
Pressurizer SRV Fails To Reseat After Relieving Liquid
TNCOSRVDEX
2
ND Recirculation Failure
CX1
Page 23

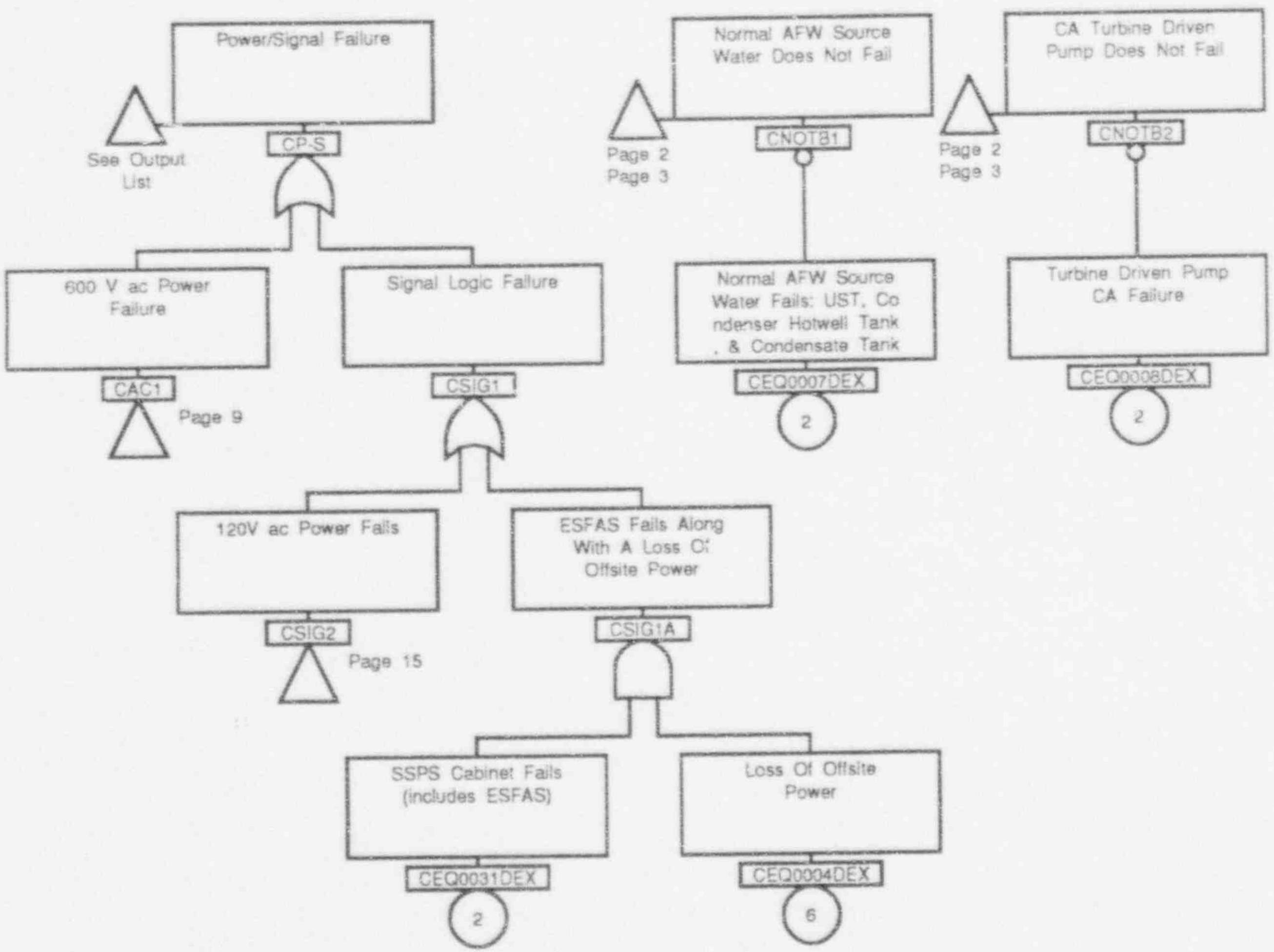
CB1 Outputs:
Page 4, Page 5, Page 5, Page 2, Page 2, Page 4, Page 3

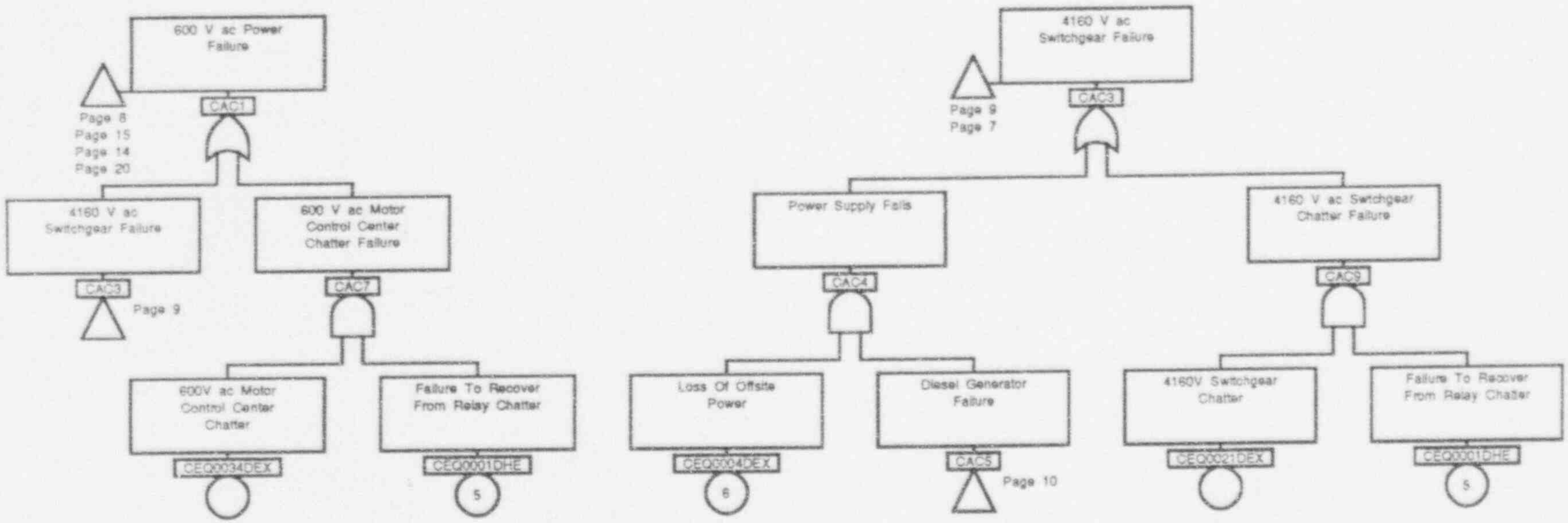


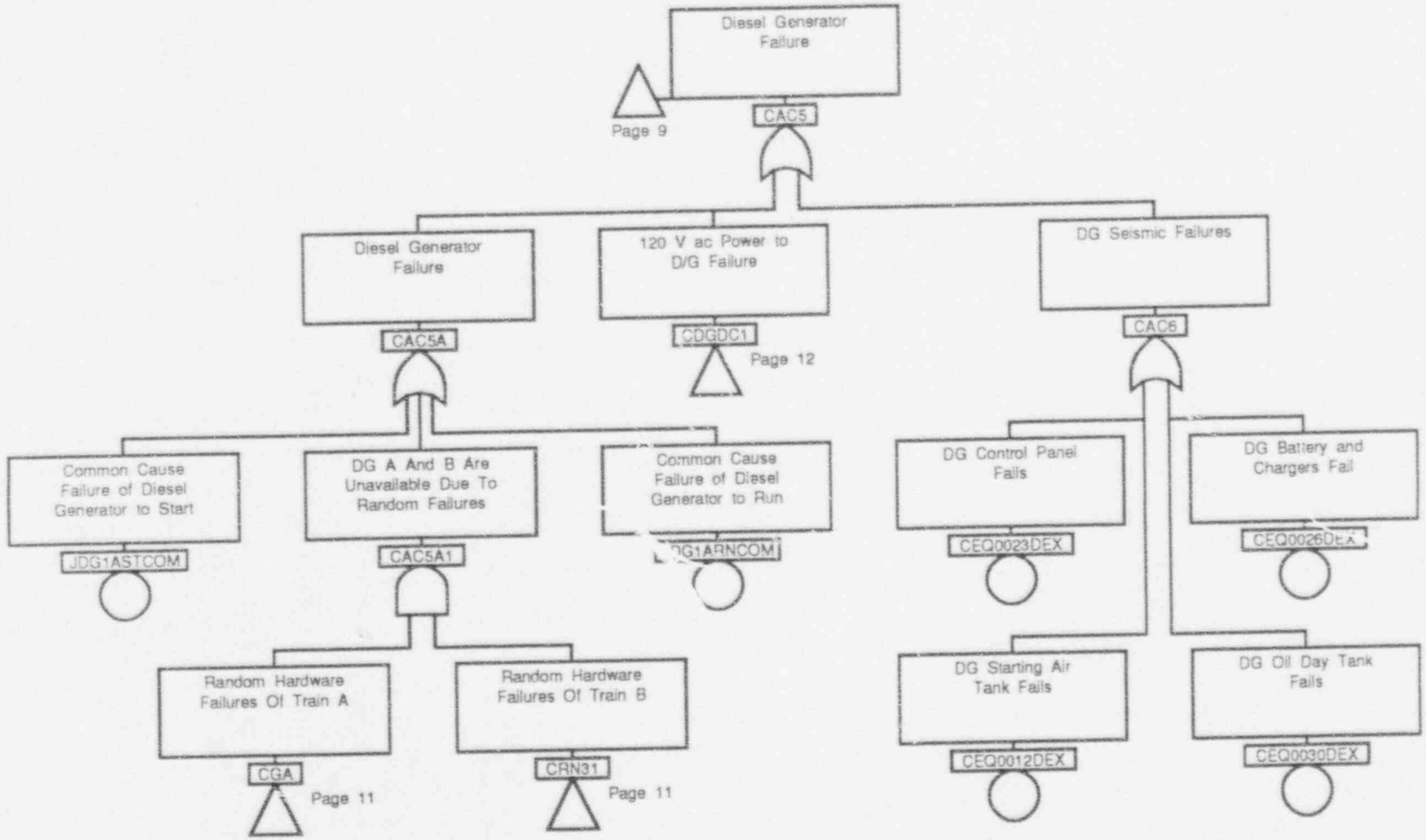
CRN1 Outputs:
Page 24, Page 22, Page 19, Page 26, Page 21, CZ1



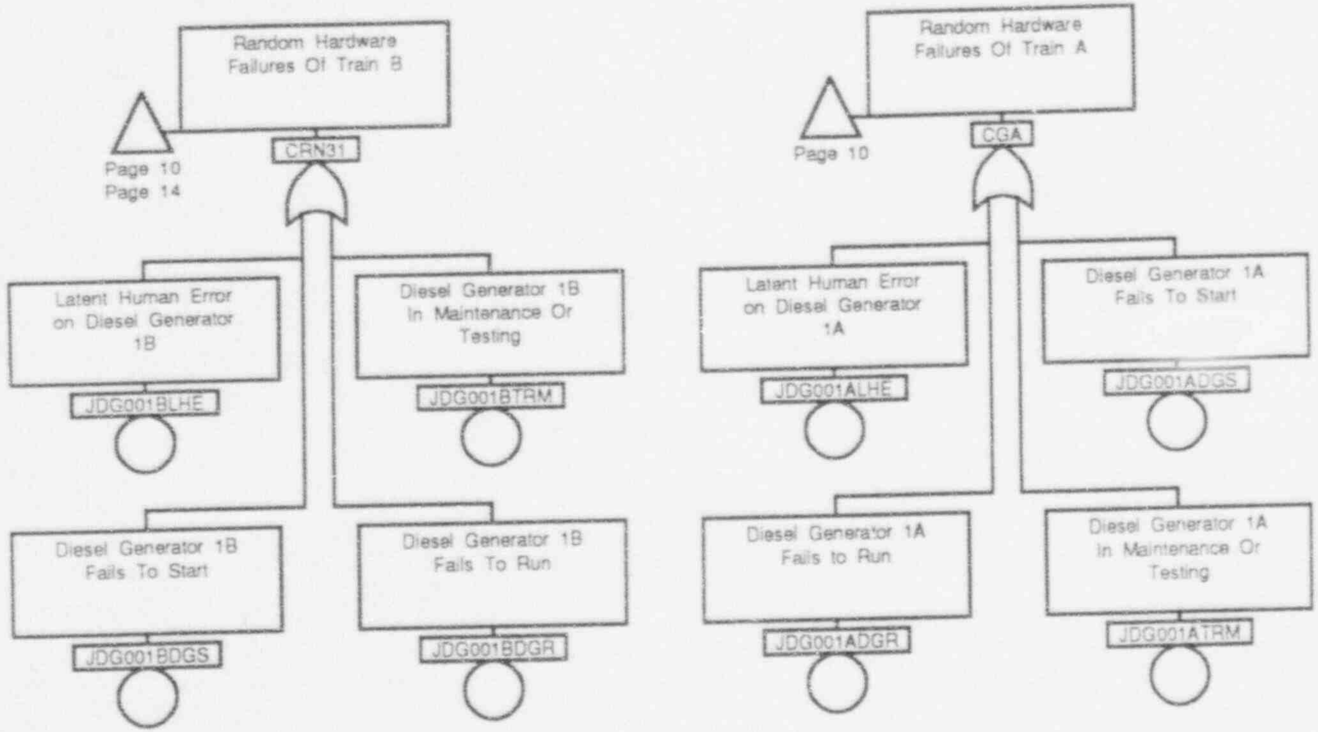
CP-S Outputs:
Page 24, Page 22, Page 7, Page 21, Page 23, CZ1, CI, CF

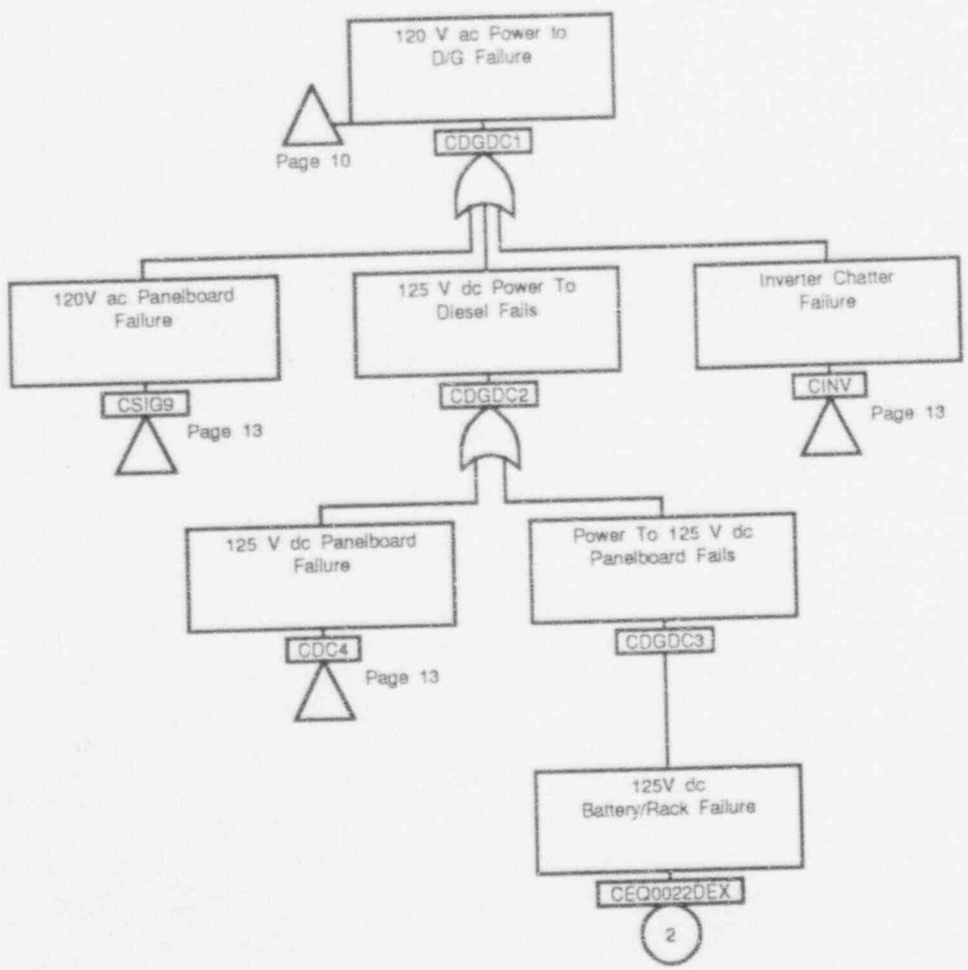


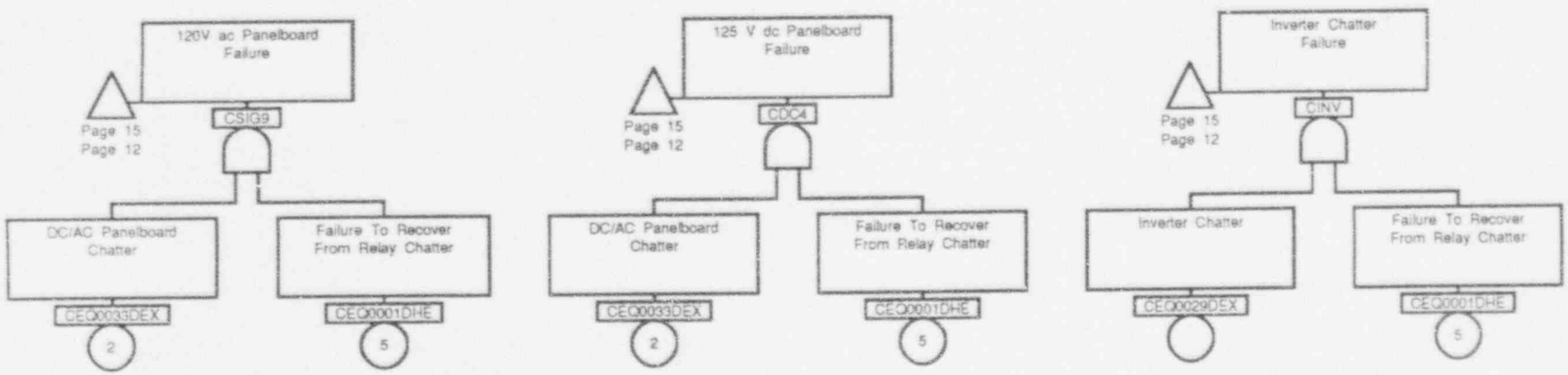

600 V ac Power Failure
CAC1
Page 8
Page 15
Page 14
Page 20
4160 V ac Switchgear Failure
CAC3
Page 9
600 V ac Motor Control Center Chatter Failure
CAC7
600V ac Motor Control Center Chatter
CEQ0034DEX
Failure To Recover From Relay Chatter
CEQ0001DHE
5
4160 V ac Switchgear Failure
CAC3
Page 9
Page 7
Power Supply Fails
CAC4
Loss Of Offsite Power
CEQ0004DEX
6
Diesel Generator Failure
CAC5
Page 10
4160 V ac Switchgear Chatter Failure
CAC9
4160V Switchgear Chatter
CEQ0021DEX
Failure To Recover From Relay Chatter
CEQ0001DHE
5

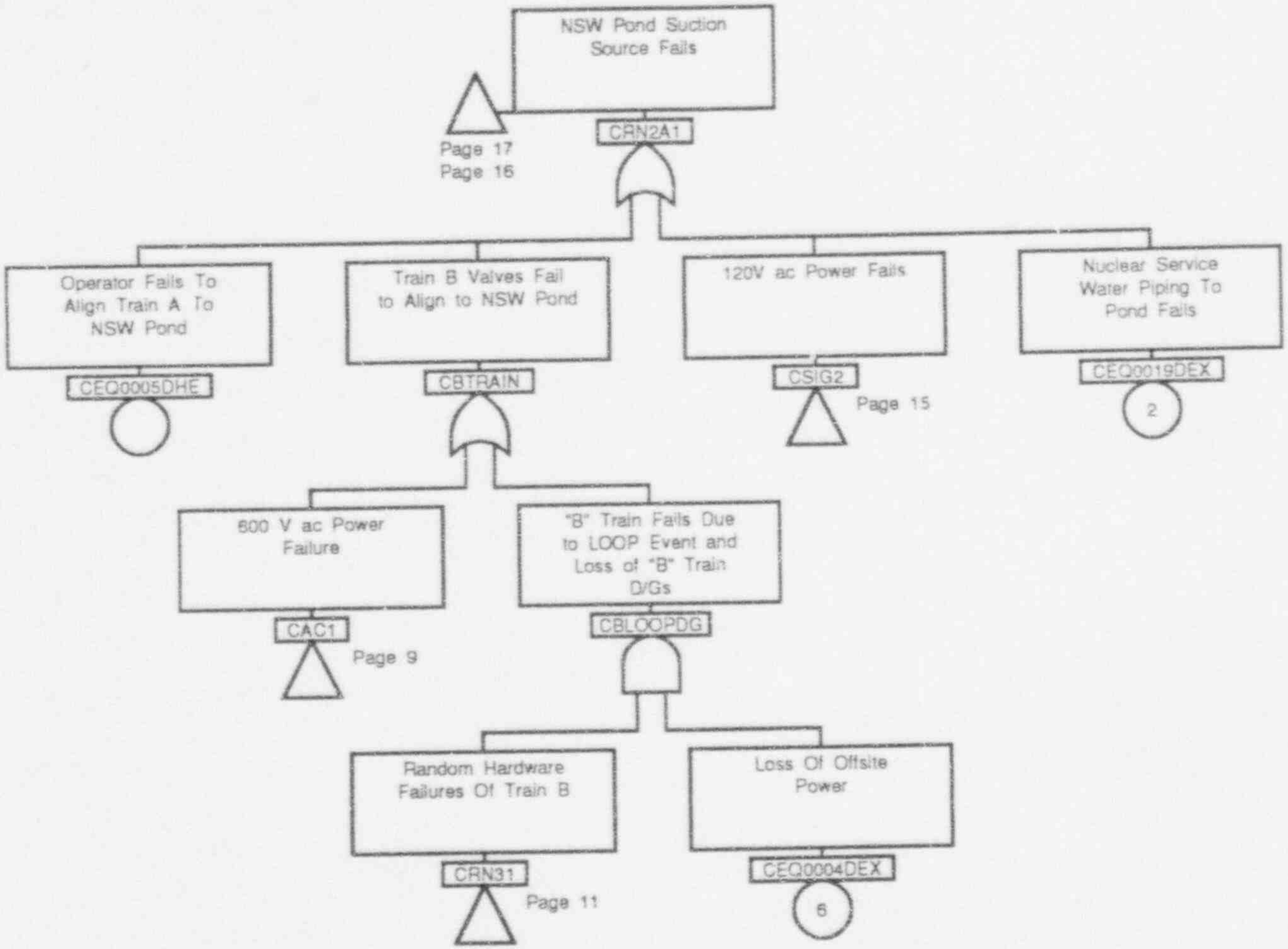


Diesel Generator Failure
Page 9
CAC5
Diesel Generator Failure
CAC5A
120 V ac Power to D/G Failure
CDGDC1
Page 12
DG Seismic Failures
CAC6
Common Cause Failure of Diesel Generator to Start
JDG1ASTCOM
DG A And B Are Unavailable Due To Random Failures
CAC5A1
Common Cause Failure of Diesel Generator to Run
JDG1ARNCOM
DG Control Panel Fails
CEQ0023DEX
DG Battery and Chargers Fail
CEQ0026DEX
Random Hardware Failures Of Train A
CGA
Page 11
Random Hardware Failures Of Train B
CRN31
Page 11
DG Starting Air Tank Fails
CEQ0012DEX
DG Oil Day Tank Fails
CEQ0030DEX

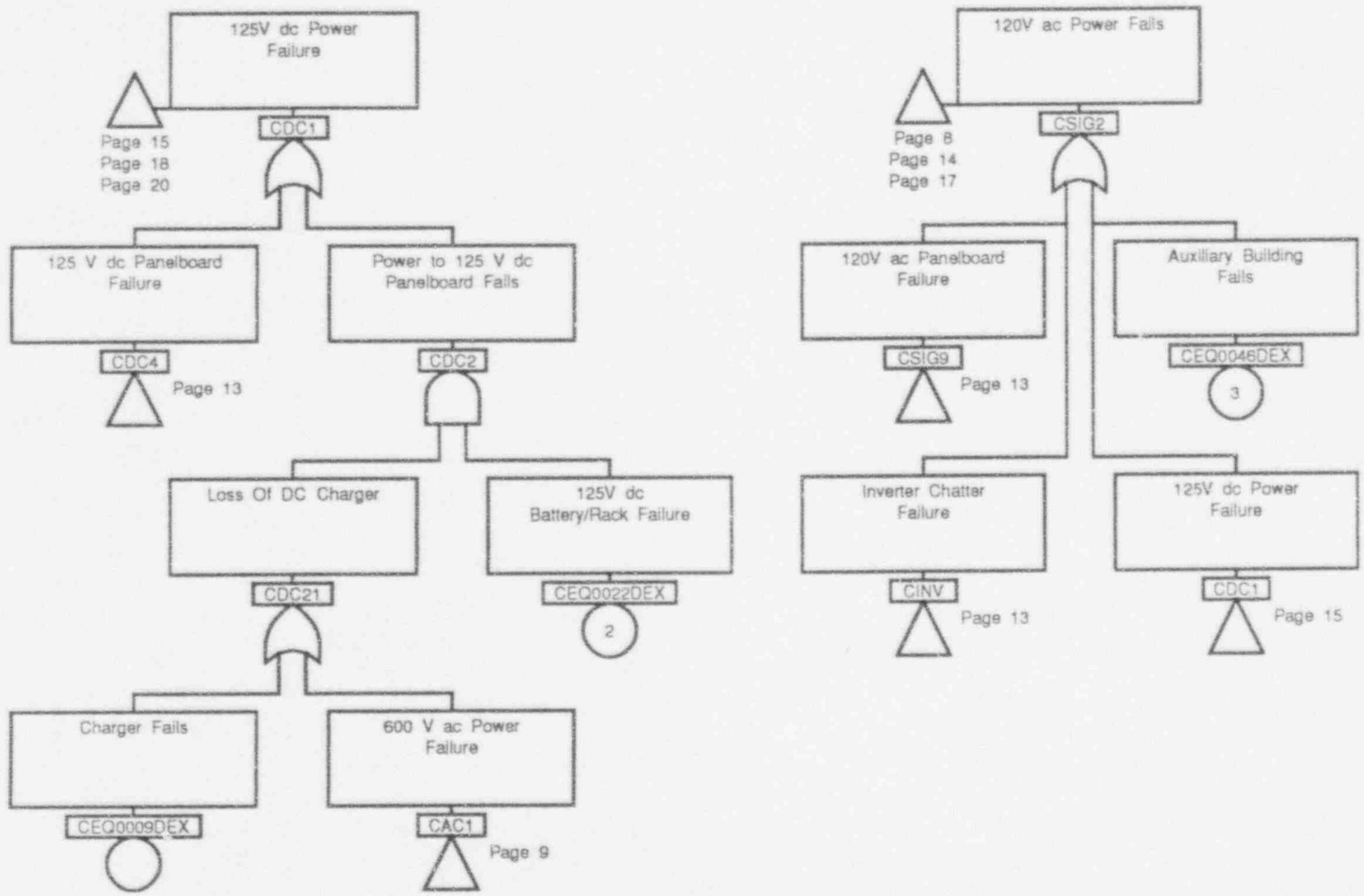

Random Hardware Failures Of Train B
Page 10
Page 14
CRN31
Latent Human Error on Diesel Generator 1B
JDG001BLHE
Diesel Generator 1B In Maintenance Or Testing
JDG001BTRM
Diesel Generator 1B Fails To Start
JDG001BDGS
Diesel Generator 1B Fails To Run
JDG001BDGR
Random Hardware Failures Of Train A
Page 10
CGA
Latent Human Error on Diesel Generator 1A
JDG001ALHE
Diesel Generator 1A Fails To Start
JDG001ADGS
Diesel Generator 1A Fails to Run
JDG001ADGR
Diesel Generator 1A In Maintenance Or Testing
JDG001ATRM

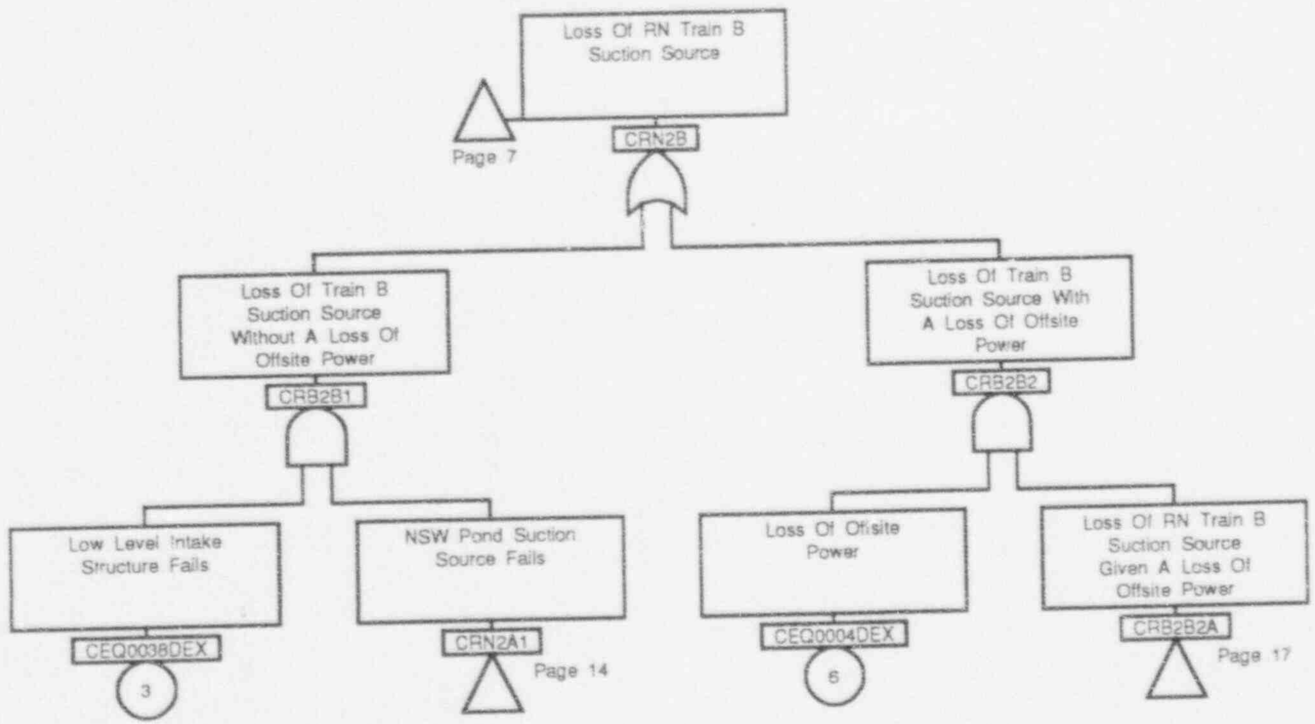

120 V ac Power to D/G Failure
Page 10
CDGDC1
120V ac Panelboard Failure
CSIG9
Page 13
125 V dc Power To Diesel Fails
CDGDC2
Inverter Chatter Failure
CINV
Page 13
125 V dc Panelboard Failure
CDC4
Page 13
Power To 125 V dc Panelboard Fails
CDGDC3
125V dc Battery/Rack Failure
CEQ0022DEX
2


120V ac Panelboard Failure
Page 15
Page 12
CSIG9
DC/AC Panelboard Chatter
CEQ0033DEX
2
Failure To Recover From Relay Chatter
CEQ0001DHE
5
125 V dc Panelboard Failure
Page 15
Page 12
CDC4
DC/AC Panelboard Chatter
CEQ0033DEX
2
Failure To Recover From Relay Chatter
CEQ0001DHE
5
Inverter Chatter Failure
Page 15
Page 12
CINV
Inverter Chatter
CEQ0029DEX
Failure To Recover From Relay Chatter
CEQ0001DHE
5

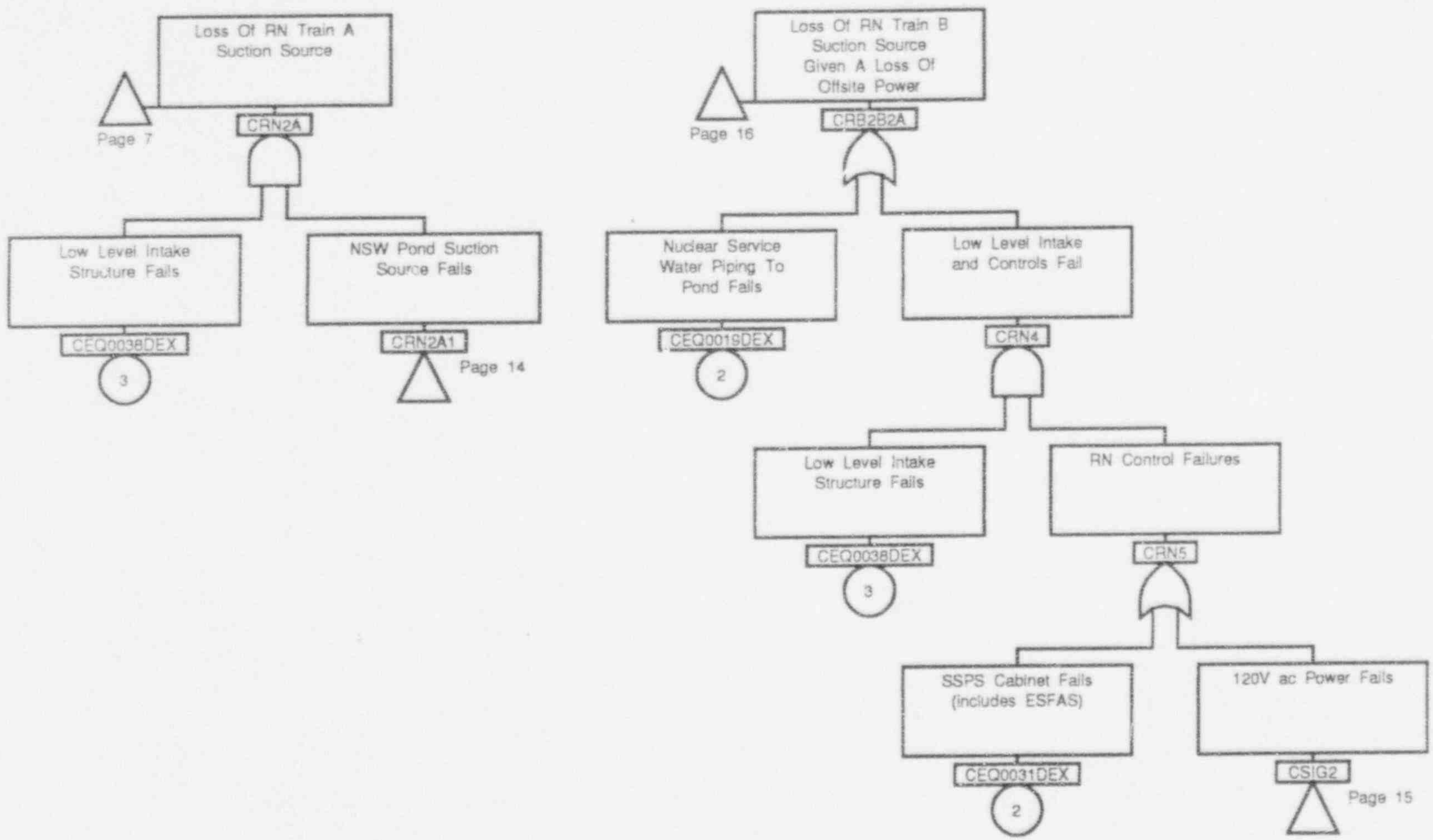


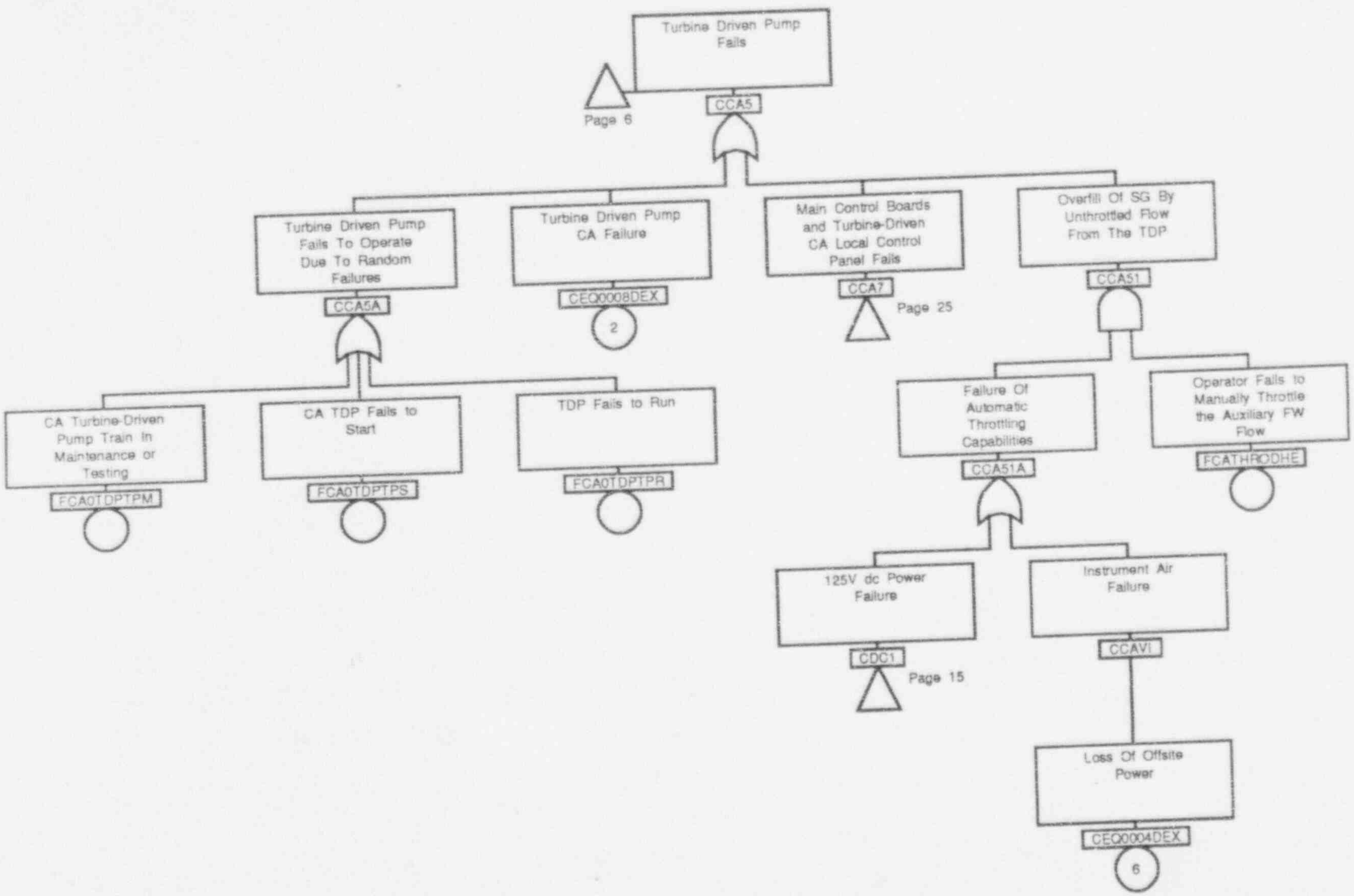
NSW Pond Suction Source Fails
CRN2A1
Page 17
Page 16
Operator Fails To Align Train A To NSW Pond
CEQ0005DHE
Train B Valves Fail to Align to NSW Pond
CBTRAIN
120V ac Power Fails
CSIG2
Page 15
Nuclear Service Water Piping To Pond Fails
CEQ0019DEX
2
600 V ac Power Failure
CAC1
Page 9
"B" Train Fails Due to LOOP Event and Loss of "B" Train D/Gs
CBLOOPDG
Random Hardware Failures Of Train B
CRN31
Page 11
Loss Of Offsite Power
CEQ0004DEX
6

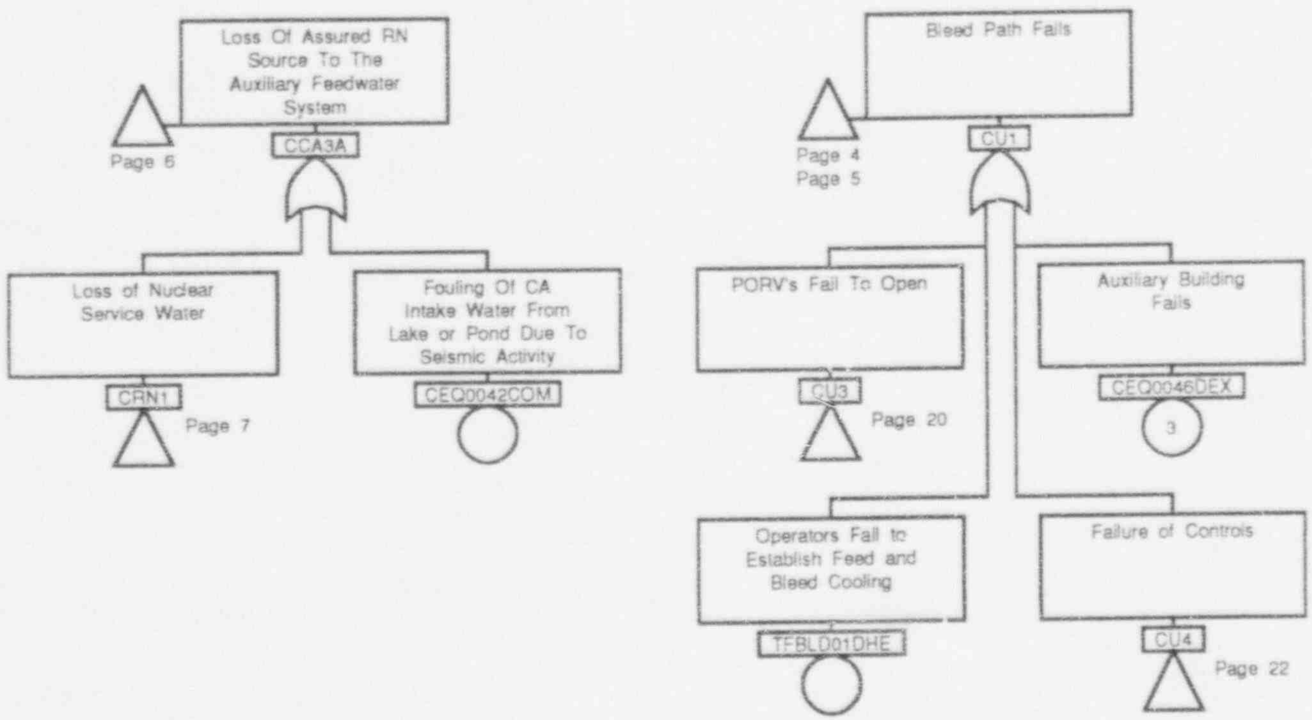

125V dc Power Failure
CDC1
Page 15
Page 18
Page 20
125 V dc Panelboard Failure
CDC4
Page 13
Power to 125 V dc Panelboard Fails
CDC2
Loss Of DC Charger
CDC21
125V dc Battery/Rack Failure
CEQ0022DEX
2
Charger Fails
CEQ0009DEX
600 V ac Power Failure
CAC1
Page 9
120V ac Power Fails
CSIG2
Page 8
Page 14
Page 17
120V ac Panelboard Failure
CSIG9
Page 13
Auxiliary Building Fails
CEQ0046DEX
3
Inverter Chatter Failure
CINV
Page 13
125V dc Power Failure
CDC1
Page 15

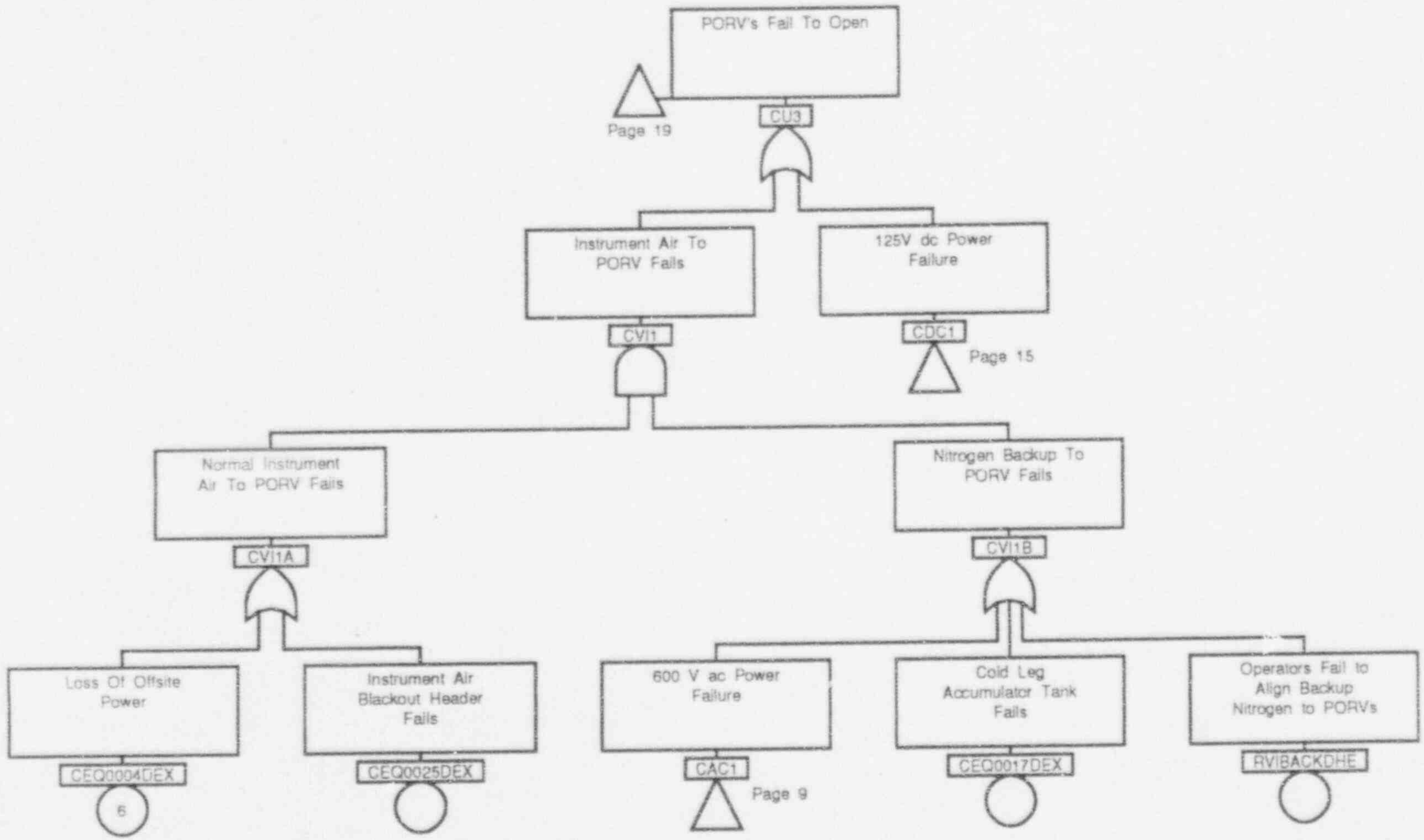


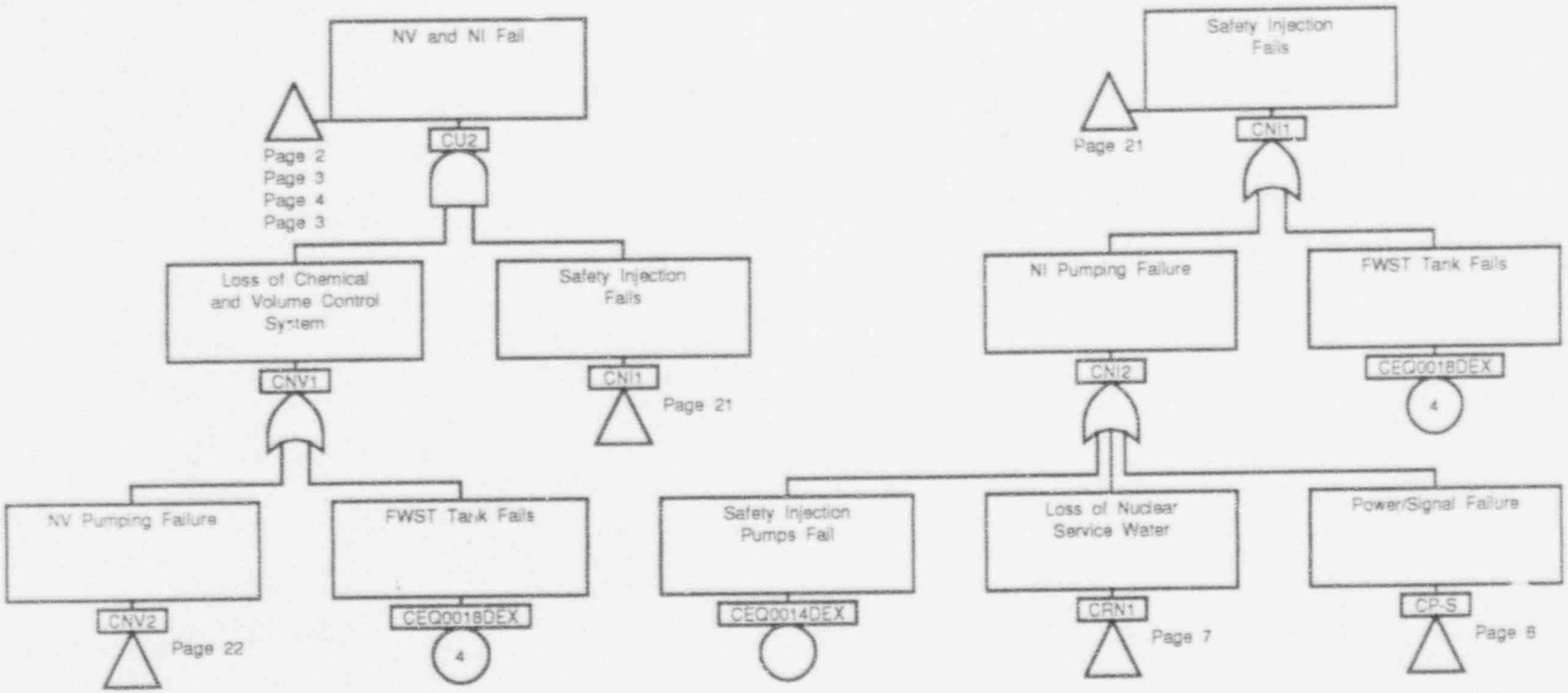
Loss Of RN Train B Suction Source
CRN2B
Page 7
Loss Of Train B Suction Source Without A Loss Of Offsite Power
CRB2B1
Loss Of Train B Suction Source With A Loss Of Offsite Power
CRB2B2
Low Level Intake Structure Fails
CEQ0038DEX
3
NSW Pond Suction Source Fails
CRN2A1
Page 14
Loss Of Offsite Power
CEQ0004DEX
6
Loss Of RN Train B Suction Source Given A Loss Of Offsite Power
CRB2B2A
Page 17


Loss Of RN Train A Suction Source
Page 7
CRN2A
Low Level Intake Structure Fails
CEQ0038DEX
3
NSW Pond Suction Source Fails
CRN2A1
Page 14
Loss Of RN Train B Suction Source Given A Loss Of Offsite Power
Page 16
CRB2B2A
Nuclear Service Water Piping To Pond Fails
CEQ0019DEX
2
Low Level Intake and Controls Fail
CRN4
Low Level Intake Structure Fails
CEQ0038DEX
3
RN Control Failures
CRN5
SSPS Cabinet Fails (includes ESFAS)
CEQ0031DEX
2
120V ac Power Fails
CSIG2
Page 15

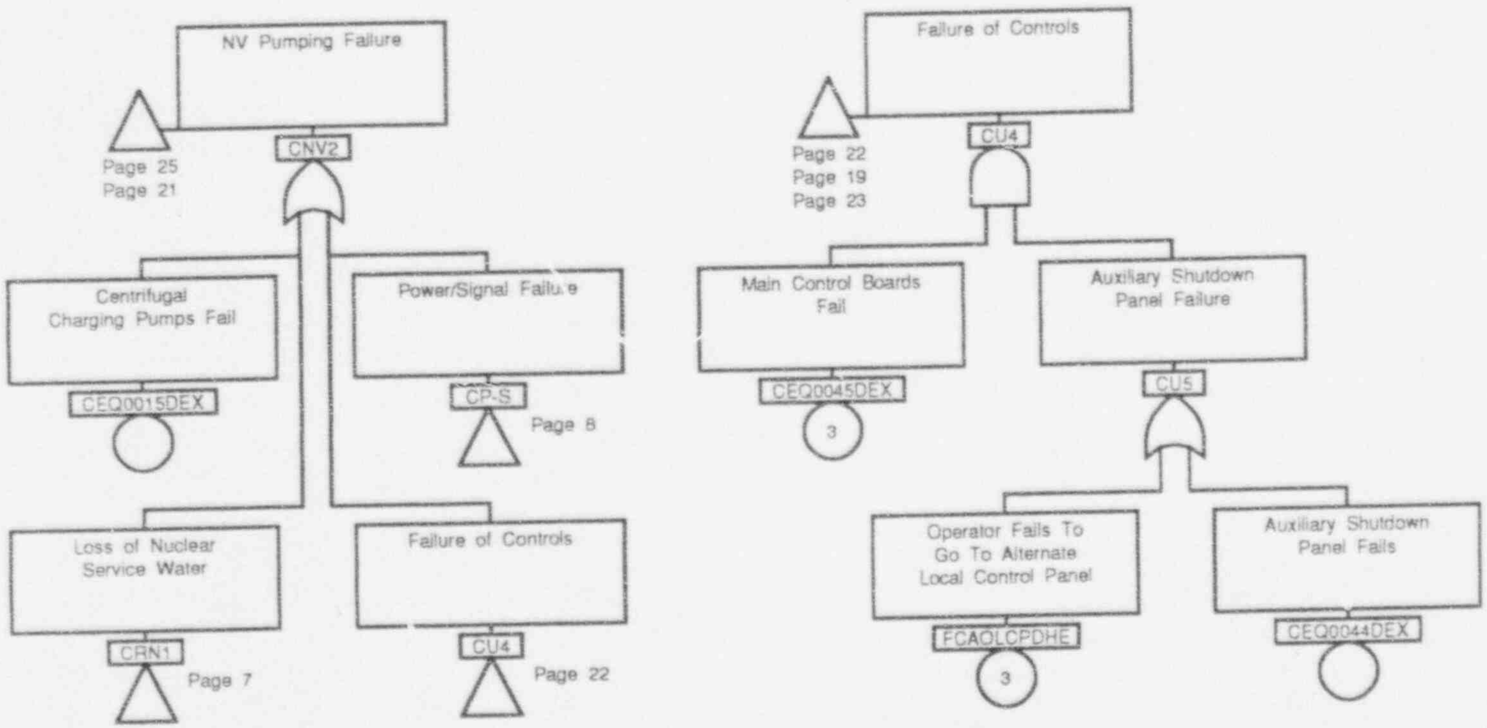

Turbine Driven Pump Fails
CCA5
Page 6
Turbine Driven Pump Fails To Operate Due To Random Failures
CCA5A
Turbine Driven Pump CA Failure
CEQ0008DEX
2
Main Control Boards and Turbine-Driven CA Local Control Panel Fails
CCA7
Page 25
Overfill Of SG By Unthrottled Flow From The TDP
CCA51
CA Turbine-Driven Pump Train In Maintenance or Testing
FCA0TDPTPM
CA TDP Fails to Start
FCA0TDPTPS
TDP Fails to Run
FCA0TDPTPR
Failure Of Automatic Throttling Capabilities
CCA51A
Operator Fails to Manually Throttle the Auxiliary FW Flow
FCATHRODHE
125V dc Power Failure
CDC1
Page 15
Instrument Air Failure
CCAVI
Loss Of Offsite Power
CEQ0004DEX
6

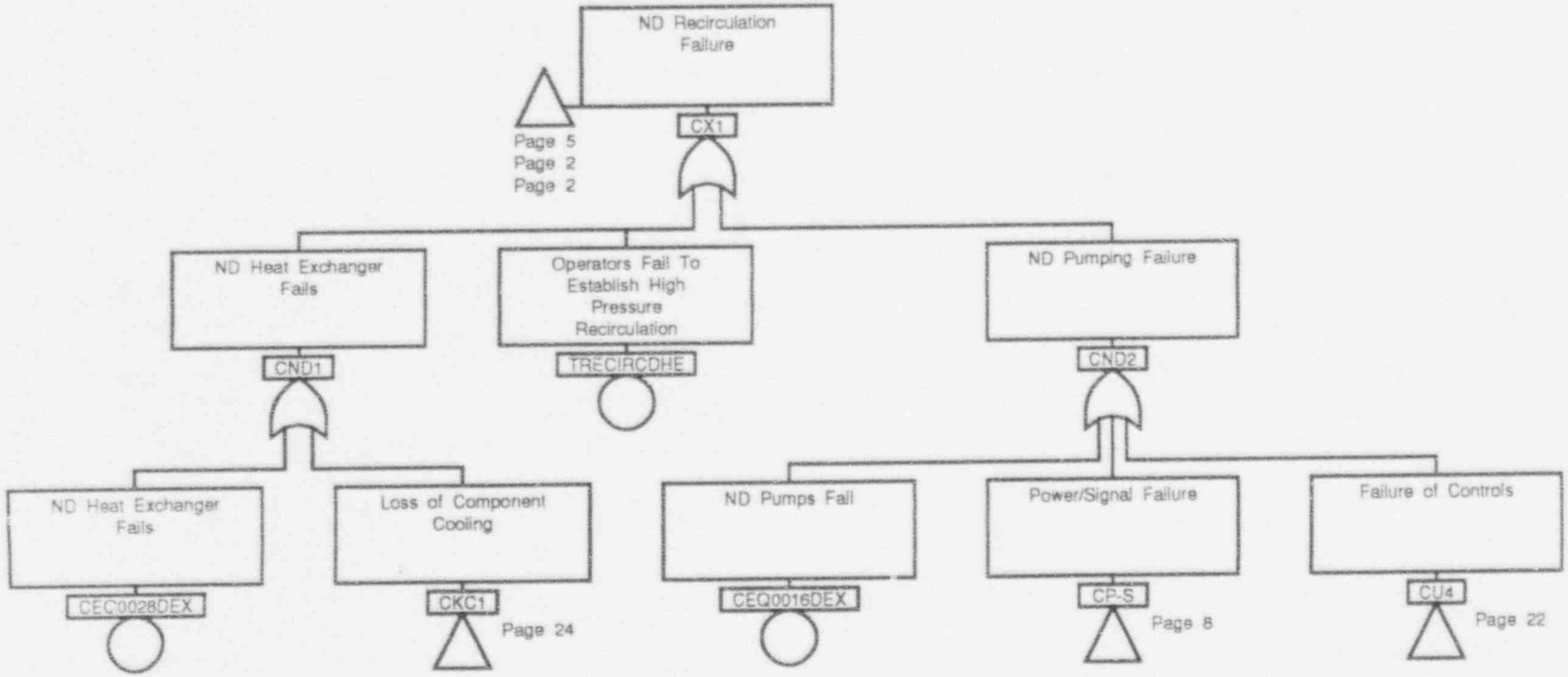

Loss Of Assured RN Source To The Auxiliary Feedwater System
CCA3A
Page 6
Loss of Nuclear Service Water
CRN1
Page 7
Fouling Of CA Intake Water From Lake or Pond Due To Seismic Activity
CEQ0042COM
Bleed Path Fails
CU1
Page 4
Page 5
PORV's Fail To Open
CU3
Page 20
Auxiliary Building Fails
CEQ0046DEX
3
Operators Fail to Establish Feed and Bleed Cooling
TFBLD01DHE
Failure of Controls
CU4
Page 22


PORV's Fail To Open
CU3
Page 19
Instrument Air To PORV Fails
CVI1
125V dc Power Failure
CDC1
Page 15
Normal Instrument Air To PORV Fails
CVI1A
Nitrogen Backup To PORV Fails
CVI1B
Loss Of Offsite Power
CEQ0004DEX
6
Instrument Air Blackout Header Fails
CEQ0025DEX
600 V ac Power Failure
CAC1
Page 9
Cold Leg Accumulator Tank Fails
CEQ0017DEX
Operators Fail to Align Backup Nitrogen to PORVs
RVIBACKDHE

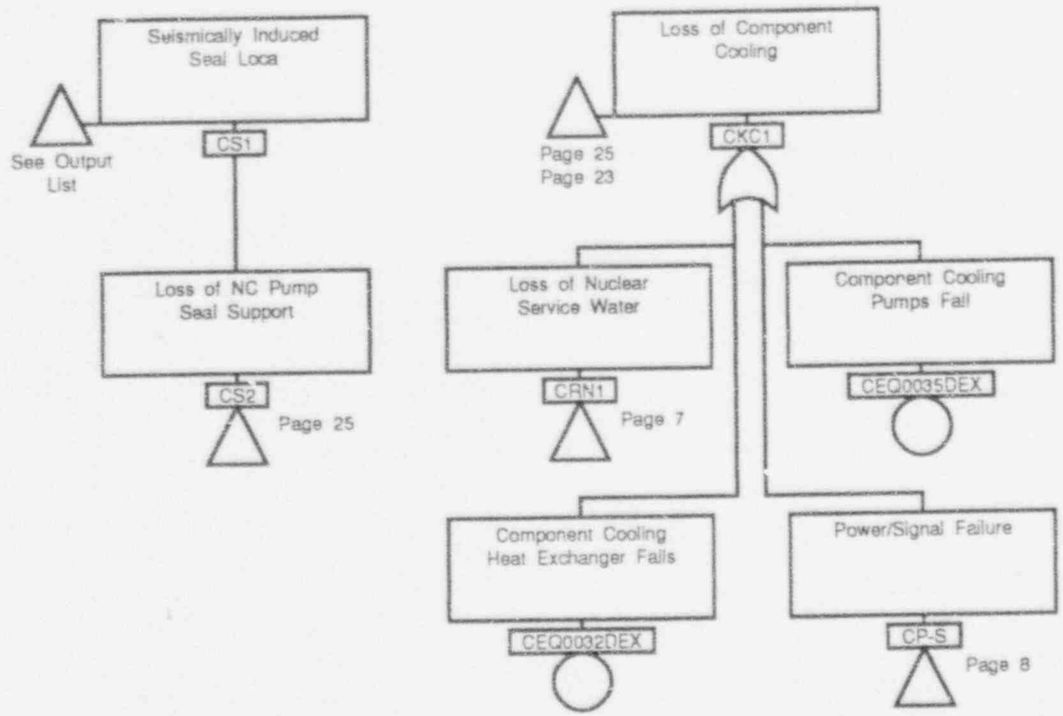


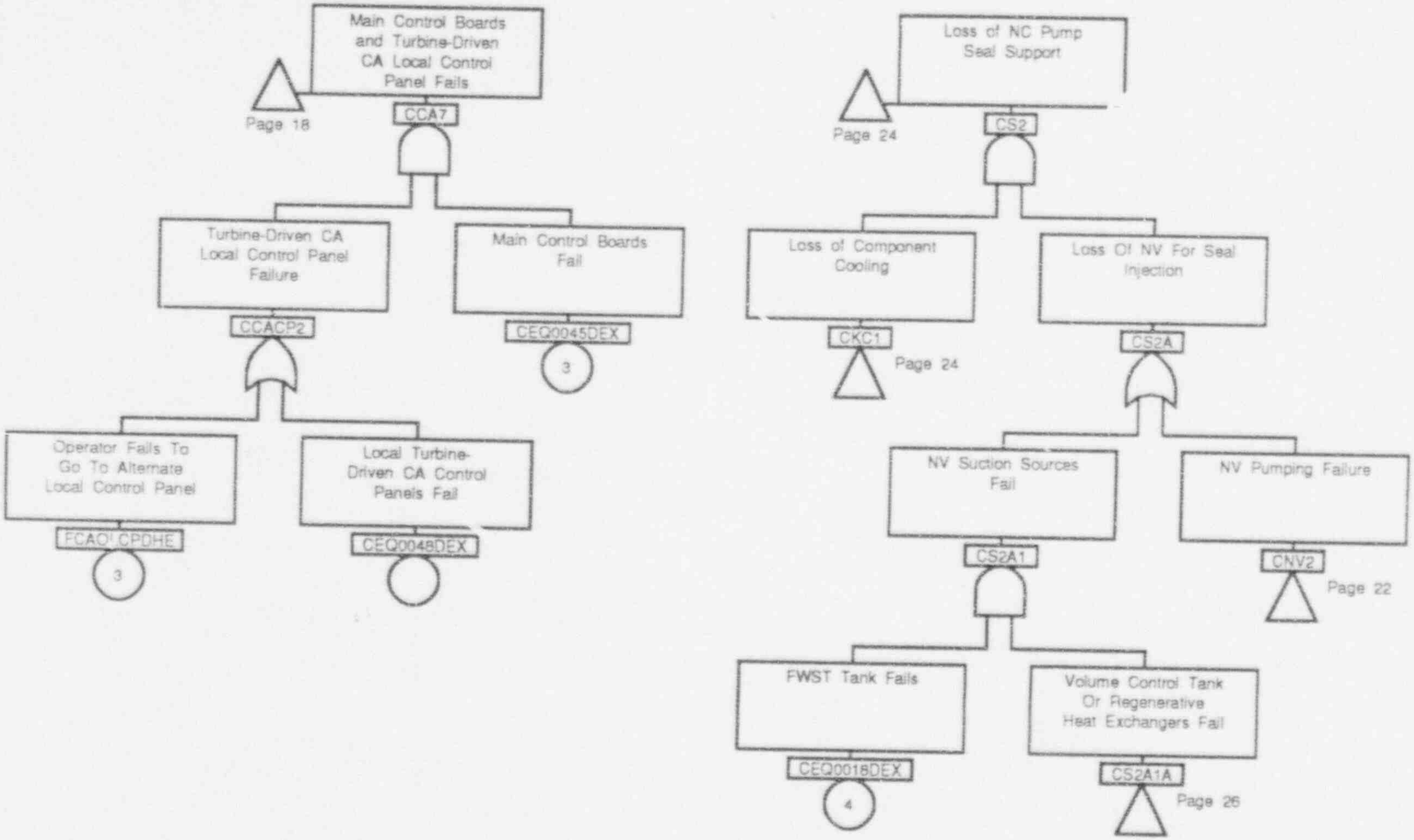
NV and NI Fail
Page 2
Page 3
Page 4
Page 3
CU2
Loss of Chemical and Volume Control System
CNV1
Safety Injection Fails
CNI1
Page 21
NV Pumping Failure
CNV2
Page 22
FWST Tank Fails
CEQ0018DEX
4
Safety Injection Fails
Page 21
CNI1
NI Pumping Failure
CNI2
FWST Tank Fails
CEQ0018DEX
4
Safety Injection Pumps Fail
CEQ0014DEX
Loss of Nuclear Service Water
CRN1
Page 7
Power/Signal Failure
CP-S
Page 8


NV Pumping Failure
CNV2
Page 25
Page 21
Centrifugal Charging Pumps Fail
CEQ0015DEX
Power/Signal Failure
CP-S
Page 8
Loss of Nuclear Service Water
CRN1
Page 7
Failure of Controls
CU4
Page 22
Failure of Controls
CU4
Page 22
Page 19
Page 23
Main Control Boards Fail
CEQ0045DEX
3
Auxiliary Shutdown Panel Failure
CU5
Operator Fails To Go To Alternate Local Control Panel
FCAOLCPDHE
3
Auxiliary Shutdown Panel Fails
CEQ0044DEX

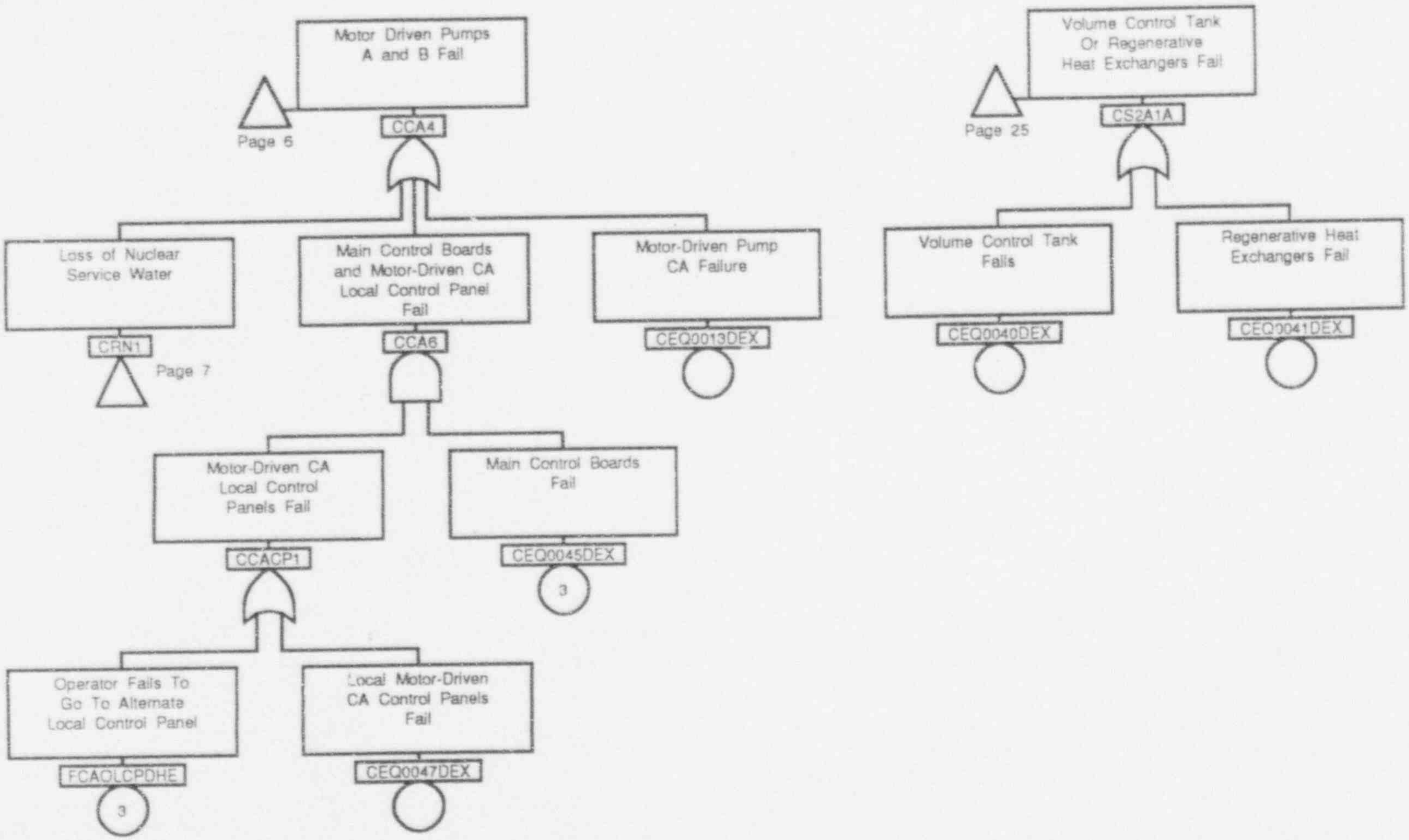

ND Recirculation Failure
CX1
Page 5
Page 2
Page 2
ND Heat Exchanger Fails
CND1
Operators Fail To Establish High Pressure Recirculation
TRECIRCDHE
ND Pumping Failure
CND2
ND Heat Exchanger Fails
CEC0028DEX
Loss of Component Cooling
CKC1
Page 24
ND Pumps Fail
CEQ0016DEX
Power/Signal Failure
CP-S
Page 8
Failure of Controls
CU4
Page 22

CS1 Outputs:
Page 5, Page 4, Page 2, Page 3, Page 4





Main Control Boards and Turbine-Driven CA Local Control Panel Fails
Page 18
CCA7
Turbine-Driven CA Local Control Panel Failure
CCACP2
Main Control Boards Fail
CEQ0045DEX
3
Operator Fails To Go To Alternate Local Control Panel
FCAO!.CPDHE
3
Local Turbine-Driven CA Control Panels Fail
CEQ0048DEX
Loss of NC Pump Seal Support
Page 24
CS2
Loss of Component Cooling
CKC1
Page 24
Loss Of NV For Seal Injection
CS2A
NV Suction Sources Fail
CS2A1
NV Pumping Failure
CNV2
Page 22
FWST Tank Fails
CEQ0018DEX
4
Volume Control Tank Or Regenerative Heat Exchangers Fail
CS2A1A
Page 26


Motor Driven Pumps A and B Fail
CCA4
Page 6
Loss of Nuclear Service Water
CRN1
Page 7
Main Control Boards and Motor-Driven CA Local Control Panel Fail
CCA6
Motor-Driven Pump CA Failure
CEQ0013DEX
Motor-Driven CA Local Control Panels Fail
CCACP1
Main Control Boards Fail
CEQ0045DEX
3
Operator Fails To Go To Alternate Local Control Panel
FCAOLCPDHE
3
Local Motor-Driven CA Control Panels Fail
CEQ0047DEX
Volume Control Tank Or Regenerative Heat Exchangers Fail
CS2A1A
Page 25
Volume Control Tank Fails
CEQ0040DEX
Regenerative Heat Exchangers Fail
CEQ0041DEX

APPENDIX B

UPDATED MCGUIRE PRA FIRE ANALYSIS

3.5 FIRE ANALYSIS

3.5.1 INTRODUCTION

Fires are identified as external hazards that contribute to overall plant risk by producing multiple component failures when propagation occurs. The purpose of the fire analysis is to evaluate the contribution of fires to the core-melt frequency at McGuire Nuclear Station.

3.5.2 METHODOLOGY

Numerous fire analysis methods were found in reviewing literature on the subject. In general, fire analyses can be divided into three categories: subjective, deterministic and probabilistic. A probabilistic method is chosen for this study because of the need to quantify risk and because it employs a methodology most consistent with that used in the balance of the PRA.

The fire analysis consisted of four steps:

1. Plant areas were analyzed for the possibility of a fire causing one or more of a predetermined set of initiating events (see Table 3.5-1).

2. If there was a potential for an initiating event to be caused by a fire in an area, then the area was analyzed for the possibility of a fire causing other events which would impact the ability to shutdown the plant. These were identified by reviewing the impact on the internal event analysis models.

3. Each area was examined with a event fire tree model to quantify fire damage probabilities.

4. Fire sequences were derived and quantified based on the fire damage probabilities and the additional failures necessary for a sequence to lead to a core melt. The additional failures were quantified by the models used in the internal events analysis (see Section 2).

Fire propagation from one fire area to another was included as part of the evaluation in step 3.

3.5.2.1 Critical Areas

A critical fire area is defined as an area where a fire has the potential to lead to one or more initiating events. Table 3.5-1 is a list of initiating events examined. Table 3.5-2 is a list of critical areas and the fire scenarios examined for each area. Table 3.5-3 lists certain fire areas which were not included in the analysis and the reasons for their exclusion. Each critical area is examined for the scenario which is thought to be the worst case result of a fire in that area. The risk from other possible scenarios is judged to be bounded by the risk from the scenarios examined.

3.5.2.2 Event Tree Development

An event tree similar to the one developed by Gallucci in his thesis (Reference 1) is used as a basis for the critical area analysis. The event tree used here is shown in Figure 3.5-1. This tree relates fire initiation, detection, suppression and propagation to equipment damage states. The stage of fire spread and damage predicted by the event tree is dependent on the scenario to which the tree is applied. In most cases, fire initiates in a particular piece of equipment, and stages of the event tree represent the spread of the fire to one and then to many adjacent pieces of equipment. For certain important electrical panels where barriers to fire spread exist, the event tree predicts the spread of fire throughout the panels.

The event tree probabilities were assigned so that the event tree results corresponded to those in a preliminary survey of fire data. Eighty fires (Reference 2, 3) occurring in PWR and BWR plants operating at power were examined. Preliminary values for the fire event tree were selected from NUREG/CR-0654 (Reference 4). Appropriate propagation values were then selected and other values were modified as necessary to make the event tree results for 80 fires correspond approximately to the damage distribution found in the data. Final event tree parameters were based on preliminary values from NUREG/CR-0654. Use of the NUREG values allowed the event tree to be specifically modified for each critical area. Table 3.5-4 lists a description of event tree parameters. Table 3.5-5 lists specific event tree parameters and resulting spreading frequencies for each piece of equipment examined.

The use of propagation probabilities listed in Table 3.5-5 allows accounting for partial suppression and self extinguishment of fires. Partial suppression is a situation where a component is damaged by fire but suppression efforts keep the fire from spreading to the next stage. The propagation probabilities are conditional upon detection since no partial suppression can occur if a fire is not detected. In most cases, the propagation values used for each fire scenario were determined from an examination of the data as described above. For fires in the essential switchgear, control room, cable room and auxiliary shutdown panel, lower propagation values were used. The lower values used in these cases reflect fire protection measures built in to the design of cables and control panels. For fires which involve lube oil, higher propagation values are used to reflect the greater chance of these fires spreading.

3.5.2.3 Initiating Frequency

The initiating frequency of a fire in a component was found to be the largest contributor to room damage frequency in most cases. Where possible, the initiation frequency is assigned the value of the number of observed occurrences in reactors divided by the number of reactor years at power. Where this was not possible, the initiating frequency was found by one of the following methods:

A. Component fire initiation frequency was partitioned from similar component initiation frequencies as follows:

$$\text{Initiation Frequency} = \text{Similar Component Frequency} \times \frac{\text{\% operating time}}{\text{\% operating time of similar component}}$$

B. The frequency of initiation in switchgear was partitioned to get the initiation frequency in certain electrical components as follows:

$$\text{Fire frequency in an electrical panel} = \text{Fire frequency in switchgear} \div \text{estimated number of switchgear in the plant}$$

3.5.2.4 Barrier Penetration

An analysis of a continuous three hour fire barrier is done for the 4160 V switchgear rooms. From reference 6 the maximum fire load for a switchgear room in 1982 was 7 lb/ft^2. From Figure 9 of Reference 5, a barrier "wearout" failure rate of 1.0E-2 (for a 3-hour fire barrier) corresponds to a fire load of about 17 lb/ft^2. Assuming that the switchgear room fire load has not increased by more than about a factor of 2 since the determination in reference 6, it should be conservative to use 1.0E-02 as the conditional rate for barrier wearout failure. This number is judged to be conservative enough to cover the situation of three hour barrier with a door. The values for barrier wearout for the 1.5 hour barriers between the nuclear service water (RN) and component cooling (KC) pumps are 1.0E-02 and 0.2. These are also based on the assumption that the fire load in these areas has not more than doubled since the determination for Reference 6. A fire at stage 3 (ABC in Figure 3.5-1) engulfs the space it is in. The probability of this is multiplied by the barrier wearout failure rate to find the probability for losing equipment on the other side of the barrier.

3.5.2.5 Hot Shorts

A hot short is a short circuit in a control cable caused by a fire. In certain cases, hot shorts may lead to loss or spurious operation of valves or equipment. For both Catawba units, hot shorts have to some extent been considered in the Fire Protection Safe Shutdown Review (Reference 5). An examination of this analysis shows that, for most areas of the plant, the effects of hot shorts can be mitigated by the use of standby equipment. An example of this is that the effects of hot shorts associated with fires in the auxiliary shutdown panel can be mitigated by the use of the Standby Shutdown Facility (SSF).

Fires in the cable room or control room are assumed to be capable of deenergizing safety-related equipment or causing valves to transfer due to hot shorts. Based on the Safe Shutdown Review Analysis, hot shorts in other areas are not considered here.

3.5.2.6 Other Fire Effects

In addition to other fire effects considered, a fire may also disable equipment through smoke damage or damage from fire suppression systems. A review of fire data and of reports of fire suppression system damage (References 5 and 7) led to the conclusion that the risk from suppression system actuation is not significant when compared with the risk from the heat of any given fire. The ventilation system air handling units for the control room, cable room and battery room are interlocked with the Fire Protection system to shut down on high smoke level. A special smoke purge exhaust fan is provided for the control room. Smoke effects in other plant areas should pose much less of a risk to plant safety systems than the heat effects of fire. For these reasons, smoke damage and fire suppression system effects were not considered in the analysis.

3.5.3 EVALUATION OF CRITICAL AREAS

Fire damage frequencies were found for individual pieces of equipment, for areas and for redundant trains of certain equipment. The damage frequencies were found by using the event tree parameter frequencies from Table 3.5-5 and the appropriate initiation frequencies found in Table 3.5-6, in the fire event tree (Figure 3.5-1). The total fire damage probabilities for all events analyzed are listed in Table 3.5-7.

Level 733 Auxiliary Building, Lower Switchgear Room

The analyzed fire in this area was a fire initiating in 4160 V essential switchgear 1ETB which disables it and spreads through a 3-hour fire barrier to 1ETA. This area was assumed to be unattended for this analysis. It is supplied with CO2 fire extinguishers, a hose station and ionization smoke detectors (see Reference 8). Since the switchgear breakers are in separate enclosures propagation of a fire in this area was judged to be less likely than average and half of the normal propagation numbers were used. A 3-hour fire barrier separates 1ETB from the adjacent train switchgear 1ETA.

The results of the analysis, using the Figure 3.5-1 event tree with the parameters shown in Table 3.5-5 and the initiating frequencies shown in Table 3.5-6, showed that the frequency of a loss of 1ETB from a fire is 3.46E-07 per year. The frequency of loss of 1ETB and 1ETA was found to be 3.46E-09 per year. A fire which destroys 1ETA would also destroy the SSF transfer relays, but this would only be a problem if the control room were also unavailable.

During the walkdown of McGuire to address the "Fire Risk Scoping Study" (Reference 9) issues, water intrusion from the ETA to the ETB switchgear room was addressed as a concern. The scenario of concern is that the lower (ETB) switchgear could be disabled due to water from fire suppression efforts in the upper (ETA) switchgear room. A bounding analysis for this scenario (Duke SAAG File 202) gave a core damage frequency of 3.14E-06. However, due to the conservative nature of the bounding analysis, the actual risk from this scenario is judged to be much lower. Therefore, associated sequences have not been presented as a part of the cut set list in Appendix D.

Because of the low frequencies associated with losing 4160 V busses from fire, these scenarios are judged to be bounded by the "loss of operating 4160 V bus" (T11) initiating event analysis.

Level 716 Auxiliary Building

Room 600 and Room 600B CA MDP and TDP Rooms

Two scenarios were analyzed for the loss of the auxiliary shutdown panel (ASP) from a fire. Since this panel contains control circuits for essential 4160 V pumps, a fire here could compromise safe shutdown capability. The first scenario involves a fire which initiates in the panel itself. This panel contains a one inch thick plate between redundant train connections. Because of this, propagation between redundant trains of wiring is considered unlikely and a propagation value of 1/10 of the plant average value is used. The auxiliary feedwater (CA) motor-driven pump (MDP) room is thoroughly covered by both ionization smoke detectors and rate of rise fire detectors. The results of the analysis showed that the frequency of loss from fire of any two redundant components in the panel is 1.74E-06 per year, and that the frequency of a loss from fire of all components within the panel is 7.54E-08 per year. The second scenario analyzed involved a lube oil fire in the CA turbine-driven pump (TDP) which spreads through a 3-hour rated fire barrier and destroys the ASP. The CA TDP room is assumed to be attended most of the time that the turbine-driven pump is operated. Since a fire in the turbine is considered to be more likely when the turbine is operating, this room is considered to be attended most of the time for this analysis. This area includes photoelectric smoke detectors and fixed temperature fire detectors. This turbine is protected by a halon suppression system which has fixed temperature actuation. Fire spreading to the ASP for this scenario is predicted by an event tree which uses propagation values for a lube oil fire. The analysis showed that the frequency for loss of the ASP from a fire in the CA TDP is less than 3.96E-08 per year.

Space 649, Nuclear Services Water Pumps

The redundant trains of Nuclear Service Water (RN) pumps are separated by a 1.5 hour fire barrier. These pumps are protected by automatic sprinklers which have water flow alarms. The area is monitored by ionization smoke detectors and rate of rise fire detectors. There are CO2 extinguishers and hose stations in this area. 1.0E-02 is used for fire barrier wearout failure. The analysis showed that the frequency for loss of one train of RN from a fire initiated in an RN pump is 2.89E-05 per year. The frequency for loss of both RN trains from this fire is 3.06E-09 per year. A fire-induced loss of RN was not examined further because this was judged to be bounded by the loss of RN initiator (T9).

<u>Level 767 Auxiliary Building</u>

Room 925, Control Room

This space is attended all of the time and is monitored by smoke detectors. It has been assigned the highest detection probabilities, and propagation probabilities that are 1/10 of the plant average. The analysis showed that the frequency of a fire damaging redundant trains of equipment controlled from the control room is 1.1E-05 per year. The frequency for all-consuming fire in the control room is 5.35E-08 per year.

Room 926, Reactor Trip Switchgear

The reactor trip switchgear has two control cabinets. A fire in these cabinets can be assumed to lead to a reactor trip. The initiation frequency of 3.68E-04 per year is already much lower than the assumed frequency of an inadvertent reactor trip and this event is not examined further.

Mechanical Equipment Room 933, Control Room HVAC System

The chillers for the Control Room HVAC System use oil and are located close together in Room 933 of the Auxiliary Building. The area is equipment with smoke detectors, there are no automatic fire suppression systems associated with the chillers. The initiation frequency used (1.43E-03 per year) is based on one fire which occurred in a chiller control panel over the reactor years examined. The analysis showed that the frequency of a loss of one chiller due to a fire is 1.11E-04 per year. The frequency for a loss of both chillers is 1.40E-04 per year. This causes a loss of control room and switchgear room HVAC. The fire-induced loss of control room HVAC is judged to be bounded by the HVAC System initiating event frequency (T15).

<u>Level 750 Auxiliary Building</u>

Room 801, Cable Room

The cable room is judged to only be unattended. This is because it is a big transient area during plant operation. It is thoroughly covered with early warning smoke detectors. Because of the spacing requirements between redundant train cables, propagation in this area is taken to be 1/10 of the plant average. There is a manual fog mist system associated with this space, and it contains C02 extinguishers and fire hose stations. The analysis showed that the frequency for damage from fire of any two redundant trains from the cable room is 1.40E-05 per year. The frequency for an all-consuming fire in the cable room is 1.08E-07 per year.

Level 733 Auxiliary Building

Room 723, Component Cooling Water Pumps

Redundant trains of component cooling water (KC) pumps are separated by a 1.5 hour fire barrier. A barrier wearout probability of 0.2 is used for this barrier. These pumps are monitored by ionization smoke and fixed temperature detectors. The automatic sprinkler system associated with these pumps has alarms in the control room. The frequency for loss of KC pumps from a fire initiated in a pump is 2.89E-05 per year for one train. The frequency for loss of both KC trains is 6.12E-08 per year. Because of the low frequency, the loss of KC is bounded by the T10 analysis.

Room 701 Vital I&C Area

This area contains the 125 V dc Vital instrument and Control Power System (EPL). A fire originating in this area does not necessarily have to pass through fire barriers to disable this system, but as a practical matter, because of the arrangement of fire barriers, it is unlikely for a single fire to disable the Vital I&C system. The dominant scenario associated with this area is judged to be the loss of Nuclear Service Water scenario described below.

Cables for the Unit 1 B train of Nuclear Service Water could be damaged by the same fire that damages the 1EVDA panel board. This panel board supplies control power for the A train 4160 V breakers. The scenario of concern is a fire which would disable

Unit 1 train B of Nuclear Service Water (while it is running) and would also disable control power to the A train of 4160 V breakers. Without operator intervention, this would result in a loss of Nuclear Service Water for Unit 1. A very large battery room fire which failed the boundary to the penetration room on level 733, could damage the power cable to the SSF standby makeup pump.

This area is equipped with early warning ionization smoke detectors, it is assumed to be unattended. There are CO2 fire extinguishers and hose stations in the vicinity. The analysis for this area, using the Fault tree in Figure 3.5-1 and the initiating event frequency in Table 3.5-6, results in the loss of both trains of Nuclear Service Water at a frequency of 1.30E-04 per year. Because of the specific location in the Vital I&C area that this fire must occur in, this frequency is somewhat conservative. This scenario is examined further in section 3.5-4.

<u>Diesel Generator 1A</u>

It is assumed that the diesel generator space is attended most of the time that the diesel is operating. Since a fire in the diesel is considered to be much less likely when it is not operating, this room is considered to be attended most of the time for this analysis. The area is equipped with rate of rise and fixed temperature fire detectors. A halon suppression system is automatically actuated by a fixed temperature detector. The loss of a diesel is not an initiating event but the diesel 1A space contains control cable which could inadvertently close two main steam isolation valves if burned. The analysis showed that the frequency of losing the diesel generator 1A space from a fire initiated in the diesel is 6.60E-04 per year. The fire-induced loss of a diesel is not examined further because the probability of this event is bounded by the probability arrived at in the PRA system analysis.

Turbine Building

The feed pump turbines on the 760' level of the Turbine Building are separated by a distance of about 30 feet. Transformers 1ATC and 1ATD are located approximately 50 feet from the Unit one feed pumps. These transformers normally supply the 4160 V essential switchgear. The feed pumps are protected by a water spray system with a water flow alarm which alarms in the control room and by an ultraviolet fire detector. There are C02 extinguishers and numerous hose stations nearby. Because it is a major transient area, this area is judged to be attended 1/3 of the time. The analysis showed that the frequency for loss of one feed pump due to a fire is 2.0E-03 per year; for a loss of two feed pumps, the frequency is 2.56E-04 per year. The fire-induced loss of feedwater is bounded by the T4 initiating event frequency. The frequency for a feed pump fire which damages both feed pumps and spreads to other equipment is 3.07E-05 per year. This is assumed to disable the normal power supply for both trains of 4160 V switchgear. This scenario is discussed further in Section 3.5.4.

Level 786 (Operating Floor)

Some turbine lube oil fires at other stations have become very large and have caused extensive damage, including structural damage to the Turbine Building. If this type of fire were to occur at McGuire it could challenge the fire barrier between the Turbine Building and the switchgear room or diesel generator room in the Auxiliary Building. The power supply cable for the Unit 2 SSF standby makeup pump which passes through the Unit 1 turbine building would also be damaged. Although such fires are possible, the overall risk from these is judged to be less than that from the feed pump scenario examined above.

For this analysis, fires are assumed to be able to cause the loss of the turbine or generator. The turbine and generator are examined separately. The Turbine Lube Oil System is protected by a water spray system with water flow alarms. The generator has no automatic fire detection or suppression systems associated with it. This level of the Turbine Building is judged to be unattended. The analysis showed that the frequency for loss of the turbine generator due to a fire in the turbine is 9.2E-04 per year. The frequency for loss of the turbine generator due to a fire initiated in the generator is 1.9E-03 per year. The fire-induced loss of a turbine is not examined further because this event is judged to be bounded by the loss of load (T2) or plant trip (T1) initiating events.

Level 739 Service Building

The instrument air compressors are located on the 739 level of the Service Building. The cable for the Unit 2 SSF Standby Makeup pump also passes through the Service Building.

There are no fire barriers between redundant instrument air compressors.

There are some ionization smoke detectors (with alarms in the control rom) in the area of the reciprocal compressors.

This area has been dismissed from consideration for a fire-induced loss of instrument air due to the following considerations:

1) The success criteria for this system in the PRA allows multiple compressors to be disabled. The system will succeed with either one of three centrifugal compressors or two of three reciprocal compressors.

2) The compressors can be supplied with AC power from either Unit.

3) There should be sufficient spacing between compressors and compressor power supplies such that a very large fire would be required to disable all compressors. This was verified by a check of drawings and a walk down of the area.

4) The loss of instrument air initiating event frequency is 0.3 per year. This is a relatively high number.

Therefore, the fire induced loss of instrument air is considered to be bounded by the loss of instrument air (initiator T12) transient event analysis.

Containment

The containment contains the reactor coolant pumps, pressurizer PORVs, ND pump suction valves (ND1, ND2), the letdown line, and the reactor coolant pump seal injection and return line. A fire in any of these components could lead to an initiating event. The pressurizer PORVs are unlikely to open due to a fire-induced hot short because the valve solenoid could not get enough voltage to actuate from a short in a single cable. If a PORV inadvertently opens, it can be reclosed by deenergizing it from outside of the control room. There are no automatic valves in the portion of the seal injection lines inside containment. If the seal return line is blocked, valve NV93 would open and no seal LOCA would occur. Valves ND1B and ND2A cannot open due to a hot short inside containment when in mode 1 because power is removed from the valve operators. A fire in a reactor coolant pump could result in a plant trip. This has not been analyzed because the frequency of this event is believed to be much less than the plant trip frequency used in the systems analysis.

Reactor coolant pump motors at McGuire are the only very high voltage equipment identified during walk downs which were not seismically mounted. A seismic induced fire in a reactor coolant pump was dismissed as a concern during the walk downs because sufficient equipment would still be available for decay heat removal during this scenario.

3.5.4 TOTAL FIRE RISK

In this section, some fire scenarios have been dismissed qualitatively because they were judged to have the same effect as other events examined elsewhere in the PRA, but at a much lower probability.

Surviving sequences with a frequency greater than 1.0E-08 per year are incorporated into the cut set list in Appendix D. First though, additional credit can be taken for some sequences. The fire scenarios for which additional credit can be taken are:

1) The fire induced loss of the Control Room,
2) The fire induced loss of the Cable Room,
3) The fire induced loss of the Auxiliary Shutdown Panel,
4) Loss of Nuclear Service Water to Unit 1 from a fire in the Vital I&C battery area,
5) The fire induced loss of the Main Feedwater Pumps.

These scenarios are discussed below:

Control Room and Cable Room

A fire in the control room or cable room would have the consequences of disabling controls to important equipment. Short circuits in these controls could trip components, cause components to start or reposition or disable the control circuit without affecting the current state of the component. Because of the many components which have control cables which pass through these areas, many different core damage sequences are possible. However, for this analysis, the scenario considered is that the fire causes a loss of both trains of Nuclear Service

Water. The assumption of this scenario is considered reasonable and slightly conservative for the following reasons:

The probability of a fire affecting a second train of components is greater than the all consuming fire probability. The McGuire PRA results are more sensitive to failures of Nuclear Service Water than to other types of equipment failure.

Offsite power is considered to be lost for the Control Room and Cable Room fire sequences.

The combined probability for a fire in the Control Room or Cable Room affecting the second train of components is 2.5E-05 per year. However, it has been estimated (here) that such fires will cause control equipment to trip before the control fuses blow 20% of the time. Therefore, the probability of actually losing both trains of Nuclear Service water from a fire is 2.5E-05 x 0.2 or 5.0E-06. If operators cross-connect equipment from the unaffected Unit or take local control of circuit breakers, a core melt can be prevented. Recovery credit can be taken for cross connecting equipment or taking local control of breakers (FIREFLDREC) this is assigned a value of 5E-02. The basis for this is described in section 5. Additional credit can be taken for the Standby Shutdown Facility (SSF). The resulting sequences are listed in Table D-7 in Appendix D. The total frequency for these sequences is: 8.1E-08.

The estimation that equipment would trip 20% of the time due to a fire in the control cables is very conservative. The type of cable used at McGuire is shielded and grounded such that conductors within the armored cables are expected to short to ground rather than developing hot shorts between cables. Heat from a fire would be expected to be transmitted to conductors by cable armor which would cause a ground. The ground condition would then cause control fuses to blow.

An additional concern about a large fire in the Control Room or Cable Room is the possibility of a fire-induced LOCA. If the pressurizer PORVs were opened by a hot short, they could be failed closed by removing power. For this reason, a fire-induced PORV LOCA was not examined. An examination of other potential LOCA paths showed that valves 1ND1B and 1ND2A could not be opened by a fire during power operation because power is normally removed from the motor operators of these valves.

AUXILIARY SHUTDOWN PANEL

Both trains of Nuclear Service water could also be lost due to a fire in the Auxiliary Shutdown Panel. The frequency for a fire affecting both trains is estimated to be 1.7E-06 per year. This can be multiplied by the 0.2 probability of actually losing both trains due to a fire and the 5E-02 credit for cross connecting equipment or taking local control of breakers (FIREFLDREC). The resulting frequency of 1.7E-08 is negligible because at least 0.2 additional credit can be taken for the SSF. This additional credit puts this type of sequence below the truncation limit of 1.0E-08.

The Auxiliary Shutdown Panel could also be lost due to a fire spreading from the Auxiliary Feedwater (CA) Turbine Driven Pump. However, this frequency (about 4E-08 per year) would still be below the truncation limit when recovered by credit for actually losing both trains (0.2) and credit for cross connecting or taking local control (FIREFLDREC at 5E-02). The frequency for these sequences is negligible.

VITAL I&C BATTERY AREA

Loss of Nuclear Service Water is the scenario of concern for fire risk in the Vital I&C area at McGuire. In this scenario the running B train pump is lost when the fire affects the power cables to the pump and the A train pump cannot be started from the control room due to a loss of breaker control power.

In the event of a loss of the B train of Nuclear Service water, operators would be directed by procedure to start the A train pump. With control power unavailable to the A train pump, operators would start the pump by closing the breaker at the 1ETA switchgear. Guidance to do this is contained in procedure OP/0/A/6100/20 "Operational Guidelines Following Fire in the Auxiliary Building or Vital Area."

The loss of Nuclear Service water from a fire in the Vital I&C area has an estimated frequency of 1.30E-04 per year. This could be recovered by credit for starting the train A Nuclear Service Water Pump at 1ETA or swapping Nuclear Service Water from another unit. Recovery credit for this is estimated at 5.0E-02 (event FIREFLDREC). This type of sequence could also be recovered by credit for the Containment Ventilation Cooling Water (RV) System. This system can supply backup cooling flow to Nuclear Service Water loads. This recovery is quantified at 0.1 (see event WRNRVBKREC, described in Section 5). Additional credit can also be taken for the SSF. The resulting sequence is presented in Table D-7 in Appendix D. The frequency for this sequence is 1.3E-07.

MAIN FEEDWATER PUMPS

If the fire in the main feedwater pumps spreads to other equipment it can cause a loss of offsite power by burning the 6.9 kV/4160 V transformers for Unit 1. Power could be restored by cross connecting power from Unit 2 or using the emergency diesels. The frequency for a feedwater pump fire which spreads to other equipment is about 3.1E-05. After the feed pumps and offsite power are lost, if both diesels are lost and the turbine-driven Auxiliary Feedwater (CA) pump is lost, a core melt is assumed due to the lack of decay heat removal. From Appendix A.17, the probability of independently losing a diesel is approximately 2.7E-01. The probability of both diesels being lost is approximately 4.7E-02. The most probable failure mode for the turbine-driven CA pump is that it is in maintenance. The probability for this is 1.4E-02. The core-melt frequency from a fire in the main feedwater pumps is estimated to be:

3.1E-05/year x 4.7E-02 x 1.4E-02 or 2.0E-08 per year.

This sequence is presented as a single event Table D-7 of Appendix D.

OVERALL CORE DAMAGE FREQUENCY

The overall core damage frequency from fire at McGuire is 2.3E-07.

3.5.5 LIMITATIONS OF THE ANALYSIS

Since fire spread and damage are processes that depend on many factors, probabilities associated with these processes are difficult to predict. Any model to predict these probabilities must have numerous limitations. The model presented here is principally limited by the data base. Large fires in nuclear power plants are rare events so the available data on fire growth and the extent of fire damage is very limited. Since much of the data is from older nuclear stations which did not conform to the requirements of 10CFR50 Appendix R, it may not apply to McGuire. Often data presented was non-uniform and was incomplete. It was usually impossible to tell how big the possibility of geometric spreading of a given fire was from the documented report on the fire. A review of data specific to McGuire proved useful in directing attention to likely fire risks. However, this data was not used to update the generic data base because the fires at McGuire were generally less serious and did not result in as much damage as the ones in the data base, and this would make it difficult to update the equipment damage model.

In general, the data base reports do not mention effects of smoke damage or damage to equipment by fire suppression systems. However, this issue was addressed in a walk down (Reference 10) to address issues raised in NUREG/CR-5088.

Every possible fire scenario could not be examined here, but it is believed that the ones examined included most of the risk.

3.5.6 REFERENCES

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9. Lambright, J.A., et.al., Fire Risk Scoping Study: Investigation of Nuclear Power Fire Risk, Including Previously Unaddressed Issues, Sandia Laboratories, NUREG/CR-5088, SAND 88-0177, January 1989.

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Table 3.5-1 Rev. 2

Initiating Events Examined

Plant Trip
Loss of Offsite Power
Loss of Main Feedwater
Loss of RN
Loss of KC
Loss of 4160 V Essential Power
Loss of VC
Loss of Auxiliary Shutdown Panel
Loss of 125 V dc/120 V ac Vital Instrumentation and Control Supply
Loss of Instrument Air
Loss of Coolant Accident

Table 3.5-2 Rev. 2

Critical Fire Areas And Scenarios (Page 1 of 2)

Critical Area	Components	Scenario
Fire Area #18 & 12	Switchgear	Fire develops in lower switchgear room and spreads through a 3-hour barrier to damage the adjacent train.
Fire Area #2 & 2A	CA TDP, CA MDP rooms	Fire develops in CATDP room and spreads through the 3-hour fire barrier to damage both CA MDPs and the aux shutdown panel.
Fire Area #4	RN pumps, NI pumps Centrifugal and Reciprocal charging pumps	Fire develops in one train of nuclear service water pumps and spreads through or around a 1 1/2 hour barrier to damage the adjacent train.
Fire Area #5	Diesel Generator 1A, Cabling for MSIVs for SG 1D and 1A	Fire in diesel 1A engulfs the space and trips two steam generator isolation valves causing a transient. The fire can spread through a 3-hour barrier to the other diesel.
Fire Area #13	Nuclear Service Water Pump Cables	Fire in the Vital I&C area causes a loss of Nuclear Service Water.
Turbine Building 760 + 6	Main Feed Pumps	A lube oil fire in a feed pump involves the adjacent feed pump. The fire can spread and disable offsite power.
Turbine Building 786 + 0	Turbine or Generator	A fire initiating in either the turbine or generator causes a loss of the turbine generator and results in a plant trip.
Fire Area #25	Control Room HVAC	A fire in CR-AHU-2 spreads to CR-AHU-1 and disables both units.
Fire Area #24	Control Room	A large fire causes the loss of the control room.
Fire Area #19, (20)	Cable Room	A large fire causes the loss of the cable room or vital functions.

Table 3.5-2 Rev. 2

Critical Fire Areas And Scenarios

Critical Area	Components	Scenario
Fire Area #21	Component Cooling Pump	Fire develops in one train of component cooling and spreads through or around a 1 1/2 hour barrier to damage the adjacent train.
Containment Building	Reactor Coolant Pumps	Fire in a reactor coolant pump results in a plant trip.
Fire Area #22, (23)	Reactor Trip Switchgear	Fire in Unit 1 or Unit 2 reactor trip switchgear and trips the reactor.
Near Corridor 932B	Control Room VC/YC	A fire initiates in a control room YC chiller and spreads to the adjacent chiller.
Service Building 739 + 0	Instrument Air	A fire causes a loss of instrument air.

Table 3.5-3 Rev. 2

Reasons For Omission Of Certain Fire Areas From The Fire Analysis

Fire Area	Components	Reasons for Omission
#1 Aux Bldg 695 + 0	NS, ND pumps	Loss of NS, ND pumps is not an initiating event. ND pumps are separated by a fire wall.
#9 & 15 Aux Bldg. 733 + 0	RCP Bkrs 1A, 1B, 1C, 1D	In each space, RCP breakers are more than 80 feet apart. The probability for a fire induced loss of these breakers is judged to be overwhelmingly smaller than the breaker failure rate.
#26 Fuel Building 778 + 10	Spent Fuel	Scenarios involving spent fuel are not examined in the PRA.
#28, #30 Dog Houses	Steam Generator Relief Valves Feedwater and Auxiliary Feedwater Valves	The frequency of a loss of Main Feedwater due to a fire in the inner doghouse is considered to be negligibly small. Recoveries are possible for these areas (as described in Reference 5).

Table 3.5-4 Rev. 2

Event Tree Parameter Description

Parameter	Description and Explanation
D_1	Early detection. This is based on room attendance and automatic detection features.
S_1	Early suppression. This is based on a combination of manual and automatic suppression. It is conditional upon detection.
P_1	Early propagation. This is a number to account for partial suppression and self suppression.
P_1'	Early propagation (no detection). This only accounts for self suppression.
D_2	Late detection. Accounts for late detection of a larger fire.
S_2	Late suppression. Accounts for later suppression of a larger fire.
P_2	Late propagation. This accounts for partial suppression and self suppression of a larger fire.
P_2'	Late propagation (no detection). This accounts for self suppression of a larger fire.
S_3	Late suppression. This accounts for suppression of a large fire which is assumed to have been detected.

Table 3.5-5 Rev. 2

Event Tree Parameter Numbers And Spreading Frequencies

Component	Event Tree Parameters (For Tree Down Branches)									Fire Growth/Damage Stages		
	D_1	S_1	P_1	P_1'	D_2	S_2	P_2	P_2'	S_3	Stage 1	Stage 2	Stage 3
1ETB	0.1	0.1	0.765	0.6	0.05	0.8	0.75	0.65	0.1	1.41E-01	4.74E-02	1.88E-03
Aux Shutdown Panel	0.1	0.1	0.953	0.92	0.05	0.8	.95	0.93	0.1	1.80E-01	9.45E-03	4.10E-04
CA TDP	0.08	0.012	0.45	0.1	0.05	0.2	.05	0.0	0.1	2.79E-02	5.39E-02	9.26E-03
Feed Pump	0.1	0.488	0.45	0.1	0.01	0.2	0.05	0.0	0.1	4.72E-01	5.98E-02	7.18E-03
Control Room	0.01	0.05	0.953	0.92	0.001	0.8	0.95	0.93	0.1	5.70E-02	2.49E-03	1.25E-05
Cable Room	0.1	0.1	0.953	0.92	0.05	0.8	0.95	0.93	0.1	1.80E-01	9.79E-03	7.53E-05
Vital I&C System	0.1	0.1	0.53	0.2	0.05	0.8	0.5	0.3	0.1	1.62E-01	2.37E-02	3.98E-03
RN Pumps	0.1	0.061	0.53	0.2	0.01	0.2	0.5	0.3	0.1	1.33E-01	2.02E-02	1.61E-03
KC Pumps	0.1	0.061	0.53	0.2	0.01	0.2	0.05	0.3	0.1	1.33E-01	2.02E-02	1.61E-03

Event Tree Parameter Numbers And Spreading Frequencies

Component	Event Tree Parameters (For Tree Down Branches)									Fire Growth/Damage Stages		
	D_1	S_1	P_1	P_1'	D_2	S_2	P_2	P_2'	S_3	Stage 1	Stage 2	Stage 3
Diesel Generator 1A	0.32	.061	0.45	0.1	0.008	0.2	0.05	0.0	0.1	1.12E-01	2.23E-01	2.57E-02
Control Room HVAC	0.1	0.1	0.45	0.1	0.05	0.8	0.05	0.0	0.1	7.75E-02	9.78E-02	1.48E-02
Main Turbine	0.80	0.1	0.45	0.1	0.05	0.2	0.05	0.0	0.1	2.28E-01	5.39E-01	5.28E-02
Generator	1.0	0.1	0.45	0.1	0.05	0.8	0.05	0.0	0.1	2.71E-01	6.64E-01	6.50E-02

Table 3.5-6 Rev. 2

Fire Initiation Frequencies

Component	Initiation Frequency	Basis
1ETB	1.84E-04	Switchgear frequency partitioned.
CATDP	4.28E-04	1/10 of main feed pump frequency.
RN Pump	1.43E-03	Same as KC pump frequenc
Main Feed Pump	4.28E-03	Occurrences over reactor years.
Main Turbine	1.43E-03	Occurrences over reactor years.
Main Turbine Generator	7.13E-03	Occurrences over reactor years.
HVAC	1.43E-03	Occurrences over reactor y s.
Control Room	4.28E-03	Occurrences over reactor years.
Cable Room	1.43E-03	Occurrences over reactor years.
KC Pumps or RN Pumps	1.43E-03	Occurrences over reactor years.
Reactor Trip Switchgear	3.68E-04	Switchgear frequency partitioned.
Aux Shutdown Panel	1.84E-04	Switchgear frequency partitioned.
Diesel 1A	2.57E-02	Occurrences over reactor years.
Vital I&C Are	1.43E-03	Same value as cable room frequency.

Table 3.5-7 Rev. 2

Total Fire Damage Probabilities

Event	Probability	Disposition
A fire causes a loss or 1ETB or 1ETA.	3.5E-07	Bounded by T11 initiator frequency.
A fire causes a loss of 1ETB and 1ETA.	3.5E-09	Below the truncation limit (1.0E-08).
A fire causes a loss of one train of Nuclear Service Water (RN).	2.9E-05	Bounded by system fault tree.
An RN pump fire causes a loss of both trains of RN (both units).	3.1E-09	Bounded by T9 initiator frequency.
A fire in the Vital I&C area causes a loss of both trains of RN.	1.3E-04	Analyzed in Section 3.5.4.
A fire causes a loss of the diesel 1A space.	6.6E-04	Does not lead directly to a plant trip.
A fire causes a loss of the main turbine generator.	2.8E-03	Bounded by T1 initiator frequency.
A fire causes a loss of both main feed pumps.	2.6E-04	Bounded by T4 initiator frequency.
A fire causes a loss of both feed pumps and offsite power.	3.1E-05	Analyzed in Section 3.5.4.
A fire causes a loss of the control room.	5.4E-08	Analyzed in Section 3.5.4.
A fire causes a loss of HVAC to the control room.	1.4E-04	Bounded by T15 analysis.
A fire causes a loss of the cable room.	1.1E-07	Analyzed in Section 3.5.4
A fire causes a loss of one train of KC.	2.9E-05	Bounded by T10 initiator frequency.
A fire causes a loss of both trains of KC.	6.1E-08	Bounded by T10 initiator frequency.
A fire causes a loss of the Aux Shutdown Panel.	1.2E-07	Analyzed in Section 3.5.4
A fire causes a loss of the Vital I & C System.	3.3E-08	Bounded by T14 initiator frequency.

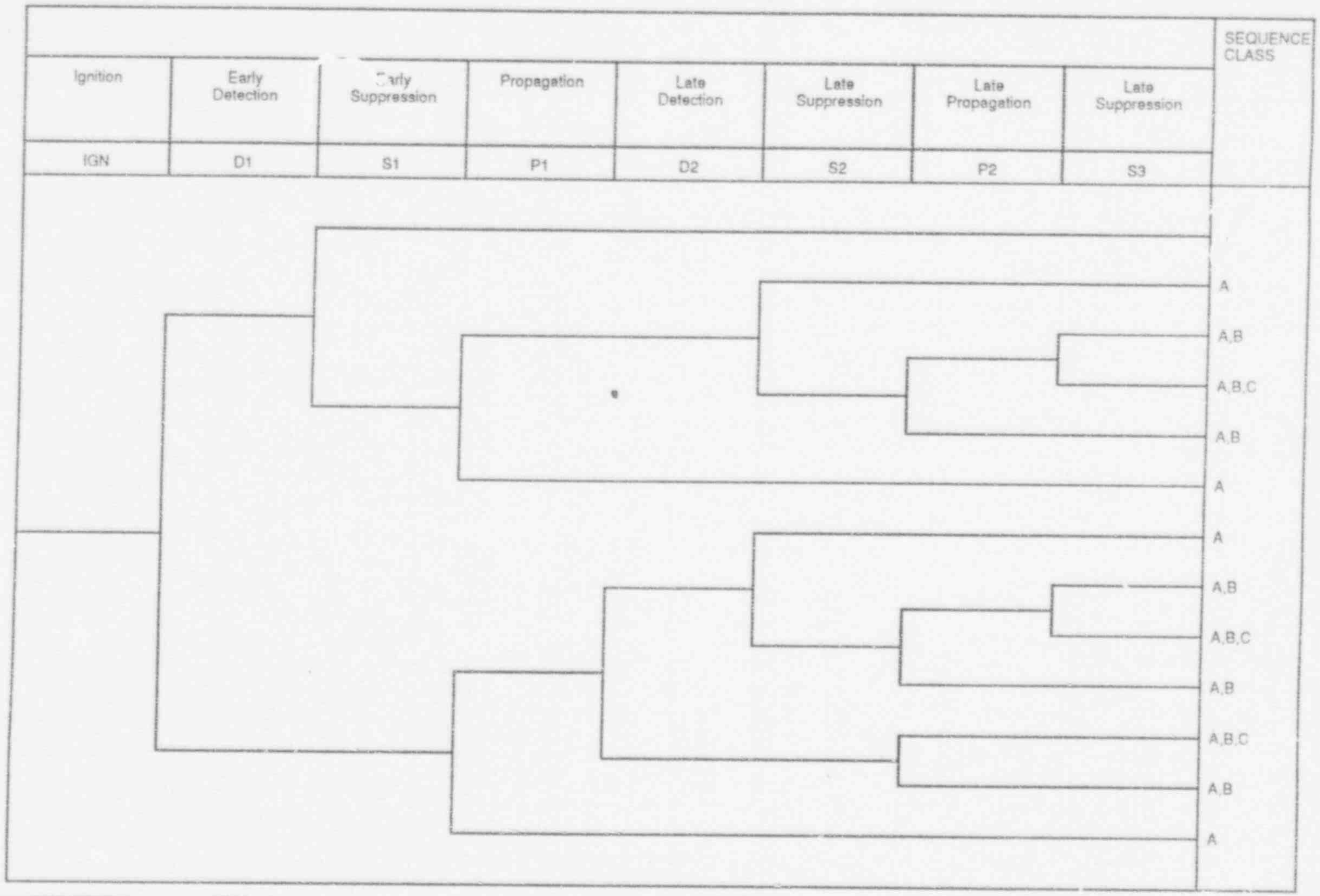

Figure 3.5-1 Rev. 2 Fire Event Tree

Table D-7 Rev. 2

Fire Event Core-Melt Cut Sets

Plant Damage State	Sequence Name	Cut Set Frequency	Percent of Total	Accident Sequence Cut Sets: Event Name	Probability	Event Description
7DI	TQSU	1.30E-07	.2	FVIC	1.30E-04	Fire in the Vital I & C Area Causes a Loss of Nuclear Service Water
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				WRNRVBKREC	1.00E-01	Operators Fail to Align RV Backup to RN
				NNVSSFADHE	2.00E-01	Operator Fails to Initiate SS Sys. Operation in Time (SSFA)
7PI	TQSU	4.25E-08	.1	CRCBFIR	5.00E-06	Fire in the Control Room or Cable Room Initiating Event
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				NSS00DGSDR	1.70E-01	SSF Diesel Generator Fails to Run
7PI	TQSU	2.50E-08	.0	CRCBFIR	5.00E-06	Fire in the Control Room or Cable Room Initiating Event
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				NNVSSFBDHE	1.00E-01	Operators Fail to Initiate SSS Operation in Time (SSFB)
7PI	TBU	2.00E-08	.0	FMFP	2.00E-08	Core Damage Sequence From a Main Feed Pump Fire
7PI	TQSU	1.30E-08	.0	CRCBFIR	5.00E-06	Fire in the Control Room or Cable Room Initiating Event
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				NSSF0DGTRM	5.20E-02	SSF Diesel Generator Unavailable Due to Maintenance
7PL	TQSU	4.25E-10	.0	CRCBFIR	5.00E-06	Fire in the Control Room or Cable Room Initiating Event
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				ZWLM221DHE	1.00E-02	Failure to Correct M221 Valves Being Left Open
				NSS00DGSDR	1.70E-01	SSF Diesel Generator Fails to Run
7PL	TQSU	2.50E-10	.0	CRCBFIR	5.00E-06	Fire in the Control Room or Cable Room Initiating Event
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				ZWLM221DHE	1.00E-02	Failure to Correct M221 Valves Being Left Open
				NNVSSFBDHE	1.00E-01	Operators Fail to Initiate SSS Operation in Time (SSFB)

Table D-7 Rev. 2

Fire Event Core-Melt Cut Sets

Plant Damage State	Sequence Name	Cut Set Frequency	Percent of Total	Accident Sequence Cut Sets: Event Name	Probability	Event Description
7PL	TBU	2.00E-10	.0	FMFP	2.00E-08	Core Damage Sequence From a Main Feed Pump Fire
				ZWLM221DHE	1.00E-02	Failure to Correct M221 Valves Being Left Open
7PL	TQSU	1.30E-10	.0	CRCBFIR	5.00E-06	Fire in the Control Room or Cable Room Initiating Event
				FIREFLDREC	5.00E-02	Operators Fail to Use Unit 2 or Remote Control During Fire
				ZWLM221DHE	1.00E-02	Failure to Correct M221 Valves Being Left Open
				NSSF0DGTRM	5.20E-02	SSF Diesel Generator Unavailable Due to Maintenance

Total: 2.32E-07 .31% of CM Total Frequency 7.4E-05