YAEC-1428

TORNADO COST-BENEFIT ANALYSIS

FOR

PROPOSED BACKFITS

AT

YANKEE NUCLEAR POWER STATION

September 1984

Principal Contributors

J. L. Staub
W. F. Lucas
S. Lee
G. A. Harper
S. M. Follen
S. P. Fournier
J. R. Chapman

Approved By: Stephen P. Schultz
Stephen P. Schultz, Manager
Nuclear Evaluation and Support Group

October 2, 1984
(Date)

Approved By: Bruce C. Slifer
Bruce C. Slifer, Director
Nuclear Engineering Department

10/2/84
(Date)

Yankee Atomic Electric Company
Nuclear Services Division
1671 Worcester Road
Framingham, Massachusetts 01701

DISCLAIMER OF RESPONSIBILITY

This document was prepared by Yankee Atomic Electric Company and is completely true and accurate to the best of our knowledge, information, and belief. It is authorized for use specifically by Yankee Atomic Electric Company and the appropriate subdivisions within the Nuclear Regulatory Commission only.

With regard to any unauthorized use whatsoever, Yankee Atomic Electric Company, and its officers, directors, agents, and employees assume no liability nor make any warranty or representation with respect to the contents of this document or to its accuracy or completeness.

ABSTRACT

A cost-benefit analysis was performed for the Yankee Nuclear Power Station to evaluate potential plant modifications aimed at reducing the risk due to tornado and wind loadings. The major modification examined involved hardening of the Safe Shutdown System to a design windspeed with an annual frequency of 10^{-5}, upper 95% confidence level. Since the results of the risk assessment performed for this analysis indicated structural failure of the Cable Tray House to be a significant risk contributor, selective hardening of the Cable Tray House was also examined.

Plant site windspeeds with annual frequencies of 10^{-4} and 10^{-5}, upper 95% confidence level, were determined to be 110 mph and 165 mph, respectively. Ultimate wind capacities were generated for all key structures and components. The risk assessment considers both hazard-induced and random failures and was performed in consonance with the PRA Procedures Guide (NUREG/CR-2300).

Justifiable costs for each backfit option were based on NRC Provisional Safety Goals, including the resource allocation basis of $1,000 per person-rem averted. Since the Yankee Nuclear Power Station, in its present configuration, meets individual and societal risk goals, plant modifications are justified only if the actual costs of the modifications are less than the calculated justifiable costs.

Results indicate that Safe Shutdown System design modifications specifically aimed at reducing the risk due to tornado and wind loadings are not justified. Upgrading the system to a design windspeed of 165 mph would cost $296,000; the ratio of actual to justifiable costs is approximately 30. Without this upgrade the Safe Shutdown System design exceeds a 110 mph design windspeed and the core melt frequency due to wind and tornado hazard is conservatively estimated to be 4.8×10^{-5}/year, upper 95% confidence level. With this upgrade, the corresponding core melt frequency is 4.1×10^{-5} which represents a 15% reduction.

ABSTRACT

(Continued)

Selective hardening of the Cable Tray House to a 110 mph design windspeed should be considered to assure availability of existing redundant instrumentation. With a modification cost of $108,000, the ratio of actual to justifiable costs is approximately 2. Since probabilistic methods do not yield an exact result, it is difficult to judge and defend with confidence that this modification is not justified. With this plant modification the core melt frequency due to the hazard is conservatively estimated to be 1.1×10^{-5}/year, upper 95% confidence level, which represents a 75% reduction. The ultimate capacity of the plant, itself, would increase to about 160 mph. The ultimate capacity of the Safe Shutdown System without modification is about 175 mph.

TABLE OF CONTENTS

		Page
	DISCLAIMER OF RESPONSIBILITY	ii
	ABSTRACT	iii
	TABLE OF CONTENTS	v
	LIST OF TABLES	viii
	LIST OF FIGURES	ix
1.0	INTRODUCTION AND SUMMARY	1
2.0	APPROACH	5
2.1	Assessment of Plant Risk (Parts 1-3)	5
2.1.1	Hazard Analysis	5
2.1.2	Plant System and Structure Response Analysis	5
2.1.3	Fragility Analysis	6
2.1.4	Plant Systems and Event Sequence Analysis	7
2.1.5	Release Frequency Analysis	7
2.1.6	Consequence Analysis	8
2.1.7	Risk Profile	9
2.1.8	Changes in Plant Risk Profile	9
2.2	Comparison of Plant Risk to Safety Goals (Part 4)	9
2.3	Cost-Benefit Analysis (Part 5)	10
3.0	PLANT MODEL DEVELOPMENT	13
3.1	Initiating Event Definition	13
3.2	Critical Safety Functions/Mitigative Systems	16
3.2.1	Reactivity Control	17
3.2.2	Main Coolant System Inventory Control	22
3.2.3	Main Coolant System Pressure Control	23
3.2.4	Core Heat Removal	24
3.2.5	Main Coolant System Heat Removal	24
3.3	Event Sequence Analysis	25
3.3.1	Loss of Off-Site Power	25
3.3.2	Excessive Cooldown	25
3.3.3	Loss of Main Coolant System Inventory	25
3.4	Identification of Critical Areas	26
3.4.1	Systems/Auxiliaries Vs. Critical Areas	26
3.4.2	Location and Description of Critical Areas	30

TABLE OF CONTENTS
(Continued)

Page

3.5 Containment Assessment 35

3.5.1 Background 35
3.5.2 Approach 35

4.0 HAZARD INFORMATION 47

4.1 Wind and Tornado Hazard Probabilities 47

5.0 FRAGILITY ANALYSIS 51

5.1 Failure Criteria 51
5.2 Structure/Component Windspeed Capacity 52

6.0 QUANTIFICATION OF PLANT MODEL 57

6.1 General Discussion of Core Melt Frequency Analysis 57

6.2 Initiating Event Frequency 58

6.2.1 Loss of Off-Site Power 58

6.2.1.1 YNPS Data 59
6.2.1.2 Industry Data 59

6.2.1.2.1 Data 59
6.2.1.2.2 Discussion of Data 61

6.2.1.3 Switchyard and Transmission Line Capacity 62
6.2.1.4 Combined Data 65

6.2.1.4.1 Tornados 65
6.2.1.4.2 High Winds 66

6.2.1.5 Results 68

6.2.2 Excessive Cooldown 68

6.2.3 Loss-of-Coolant Accidents 70

6.2.3.1 Piping/Component Physical Failures 71
6.2.3.2 Isolation Failures 71
6.2.3.3 Relief Valve Failures 73

6.3 Top Event Development 75

TABLE OF CONTENTS
(Continued)

Page

6.4 Failure Data Development 88

6.4.1 Fault Tree Basic Events 89
6.4.2 Top Events 96

6.5 Location Failure Data 99

6.6 Core Melt Quantification 107

6.6.1 Mission Time 107
6.6.2 Event Tree Quantification 108
6.6.3 Overall Results 125

6.7 Release Frequency 129

7.0 CONSEQUENCE ASSESSMENT 135

7.7.1 YNPS PSS Release Category Discussion 135
7.7.2 Use of the YNPS PSS Release Category Information 139
7.7.3 Results 139

8.0 RISK ASSESSMENT 145

9.0 COST-BENEFIT ASSESSMENT 150

9.1 Costs for Structural Upgrade 150

10.0 REFERENCES 152

APPENDIX A - Acronym Table A-1

LIST OF TABLES

Number	Title	Page
3-1	Master Logic Diagram Initiating Event Categories	37
3-2	Initiating Events	39
3-3	Critical Areas Vs. Systems	40
4-1	Tornado Hazard Probabilities	48
4-2	Straight Wind Hazard Probabilities	49
5-1	Structures/Components for Fragility Evaluation	55
5-2	Wind Capacities of Structures/Components	56
6-1	Data to Assess Relief Valve Challenge Induced LOCA	131
6-2	Logic Expression Basic Events	132
7-1	Associated Release Category Parameters Required for Consequence Calculations	142
7-2	Release Fractions - 5% Bound, 50% Confidence Level, and 95% Bound	143
7-3	Expected Conditional Individual, Societal, and Person-Rem Risk Level Values	144
8-1	Individual Acute Fatality Risk Development	146
8-2	Societal Latent Cancer Fatality Risk Development	147
8-3	Person-Rem Exposure Development	148
8-4	Risk Levels	149
8-5	Safety Goal Comparison	149
9-1	Cost-Benefit Analysis Results	151

LIST OF FIGURES

Number	Title	Page
2-1	Risk Assessment Procedure for External Events	12
3-1	Master Logic Diagram - All Events - Excessive Release	41
3-2	Master Logic Diagram - Excessive Release Due to In-Plant Events	42
3-3	Generalized Critical Safety Function Based Event Tree	43
3-4	Loss of Off-Site Power Event Tree	44
3-5	Relief Valve Challenges Event Tree	45
3-6	LOCA Event Tree	46
4-1	Hazard Curves	50
6-1	Feedwater Diagram	134

1.0 INTRODUCTION AND SUMMARY

NUREG-0825, Integrated Plant Safety Assessment, SEP, YNPS (Reference 13) Topic III-2, Wind and Tornado Loadings, describes Yankee Atomic Electric Company's (YAEC) approach to resolution of this topic. In general, that document describes three approaches which the NRC Staff would find acceptable for resolution of this topic. Under the more general topic II-2.A, "Severe Weather Phenomena", YAEC proposed to use the median value 10^{-5} wind speed as the design-basis tornado consistent with the YNPS probabilistic safety study risk levels. The NRC Staff found this approach to be generally acceptable and in accordance with option three of their proposed resolution with the following specific recommendations:

1. Determine the capability of the structures, systems and components necessary to ensure the ability to reach hot shutdown to withstand the NRC's determined 10^{-4} and 10^{-5} upper 95% confidence level wind speed,

2. Determine the plant modifications necessary to protect against both wind speeds,

3. Estimate the cost of any necessary modifications for each value of wind speed, and

4. Perform a cost/benefit analysis to support a determination of which modifications should be made.

In order to address the above, a cost-benefit analysis of potential plant modifications aimed at reducing the risk due to tornado and wind loadings was performed. The analysis consists of the following:

1. Assessment of the plant risk for the "Base Case".

2. Assessment of the plant risk as a function of each potential upgrade.

3. Assessment of the reduction in plant risk as a function of each upgrade.

4. Comparison of the "absolute" risk levels (individual and societal) to provisional safety goals (Reference 1) for the "Base Case" and each upgrade.

5. Comparison of the justifiable expenditures to reduce the residual risk to the actual costs required for each upgrade.

In the scoping of this analysis, the analysis team determined that the Safe Shutdown System (SSS) which had been committed to as an alternative to the Seismic Upgrades for the SEP, could provide redundancy to present plant systems and should be included in the "Base Case" plant analysis. Therefore, the "Base Case" risk assessment assumes that the SSS has been installed including necessary structural changes to support this upgrade.

The PRA Procedures Guide (NUREG/CR-2300, Reference 2) provides a discussion of approaches available to assess the risk due to wind and tornado loadings. The approach taken in this study to assess plant risk is in consonance with these approaches. Since both hazard-induced failures and non-hazard-induced random failures were modeled, a realistic assessment of the impact of additional plant modifications could be made.

Ultimate structural capacities of the "Base Case" structures and components were determined (presented in Section 5.0). This analysis found that the ultimate wind loading capacity of the Safe Shutdown System and related structures, as designed for seismic concerns, exceeds the capacity of the system if designed for the 10^{-4} 95% wind (110 mph). Additionally, the system capacity exceeds that of the median 10^{-5} tornado wind (also 110 mph), resolving a staff concern from SEP Topic III-2 (Reference 13). Ultimate wind loading capacity of the seismic design Safe Shutdown System is actually limited by the Upper Level Primary Auxiliary Building (ULPAB) which houses the SSS discharge header; the upper level PAB is predicted to fail at about 165 mph.

For the "Base Case", the limiting plant area was found to be the Cable Tray House. The capacity of this area is about 70 mph for either winds or tornados. Failure of this area limits Control Room instrumentation. Local readings from penetrations would remain available. Overall, the "Base Case"

core melt frequency is dominated by a combination of a 70 mph wind-induced failure of the Cable Tray House and random failure of the SSS. This eliminates instrumentation excepting local readings.

Since hazard-induced and random failures of other systems were smaller contributors, design modifications to the Cable Tray House were investigated. A proposed modification would design the Cable Tray House to a wind and tornado loading capacity of 110 mph (a 10^{-4} 95% confidence wind or 10^{-5} median tornado). Ultimate capacity of this modified area would exceed 185 mph. The plant capacity is then limited by the Steam-Driven Auxiliary Boiler Feedwater Pump Room capacity (about 160 mph ultimate). With this modification, random failures do not dominate the core melt frequency.

Modifications to the Safe Shutdown System were also considered. If the seismic design criteria for the Safe Shutdown System is upgraded by designing for the 165 mph windspeed, the SSS would ultimately withstand windspeeds exceeding 200 mph. The potential gain achievable by implementation of this modification was conservatively maximized in this evaluation by assuming a SSS capacity consistent with the predicted containment capacity of 250 mph. Core melt frequency for the modified plant, like the base case, is dominated by hazard induced failure of the Cable Tray House in combination with random SSS failure.

Section 9 provides details of the cost-benefit analysis and results. The Table below summarizes those results. Note that "Plant Capacity" does not credit the SSS; the SSS capacity is shown separately.

Description	Plant Capacity (mph)	SSS DSN Capacity (mph)	SSS Ultimate Capacity (mph)	Hazard Confidence %	Justif. Cost($)	Actual Cost($)	Ratio of Actual to Justif.
Base Case	70	-	175	-	-	-	-
Cable Tray House Upgrade	160	-	175	50	3.9K	108K	28
				95	52.5K	108K	2
SSS Upgrade	70	165	250	50	0.3K	296K	987
				95	9.9K	296K	30

As demonstrated by these results, there are no significant backfits justified from a cost-benefit perspective over and above the previously agreed

upon seismic upgrades. The reduction in person-rem as a result of hardening the Cable Spreading Room for the 10^{-4} windspeed (110 mph) is 5.3 person-rem from the baseline case with a justifiable cost of $52,500 and a modification cost of $108,000. For upgrading the baseline to incorporate the SSS at the 10^{-5} windspeed (165 mph), the reduction in person-rem is only 1.0 person-rem from the baseline case with a justifiable cost of $9,900 and a modification cost of $296,000. Based on these results, hardening of the Cable Spreading Room is the most cost-effective modification and is, therefore, more justified than upgrading the SSS to the 10^{-5} windspeed. Furthermore, the hardening of the Cable Spreading Room results in an ultimate plant capacity of approximately 160 mph without crediting the SSS.

The overall results of this evaluation support the findings of the Yankee Probabilistic Safety Study and further confirm that the plant poses an extremely small risk for those events evaluated. It is important to note that the plant residual risk is extremely low. The residual risk is low for the following reasons:

1. The frequency of core melt is low for the base case. A value of 4.8×10^{-5} per year is evaluated for the 95% confidence level hazard curve; 1.3×10^{-5} per year for the 50% confidence level hazard curve.

2. Containment integrity is assured for winds/tornados up to about 250 mph.

   Since the containment is maintained isolated regarding direct release paths, the potential for early releases is low. Additionally, containment cooling is passive. Active cooling systems are not required to preclude overpressure failure.

3. The plant core inventory is low relative to more recent designs (600 MWt versus 3000 MWt).

2.0 APPROACH

As discussed in Section 1.0, the analysis consists of five parts. The technical approach used and the information required to perform each of these five parts are discussed below.

2.1 Assessment of Plant Risk (Parts 1-3)

The PRA Procedures Guide (NUREG/CR-2300) provides a discussion of approaches available to assess the risk due to wind and tornado loadings. Figure 2-1, which is a reprint of Figure 10-1 from NUREG/CR-2300, illustrates the basic elements of this assessment. A discussion of each of these basic elements is provided below.

2.1.1 Hazard Analysis

The hazard analysis, involving an evaluation of exceedance frequency versus hazard intensity, is the driving function for the remainder of the analysis. Section 4.0 presents the hazard analysis performed in support of this study.

The hazard analysis was not performed in a complete probability of exceedance frequency and intensity manner because it was judged that the available work represented a sufficiently broad range of values for this assessment. Section 4.0 develops this basis in more detail.

Because a detailed "probability of frequency" relationship for the hazard was not deemed necessary the remainder of the analysis was based on two hazard curves, the 50% and 95% confidence level curves. Plant risk was assessed using each of these curves separately. The results, therefore, provide a reasonable quantification of the uncertainty in plant risk due to uncertainty in the hazard curves.

2.1.2 Plant System and Structure Response Analysis

As discussed in NUREG/CR-2300, "the purpose of this analysis is to translate the hazard input" into the responses on plant structures, piping

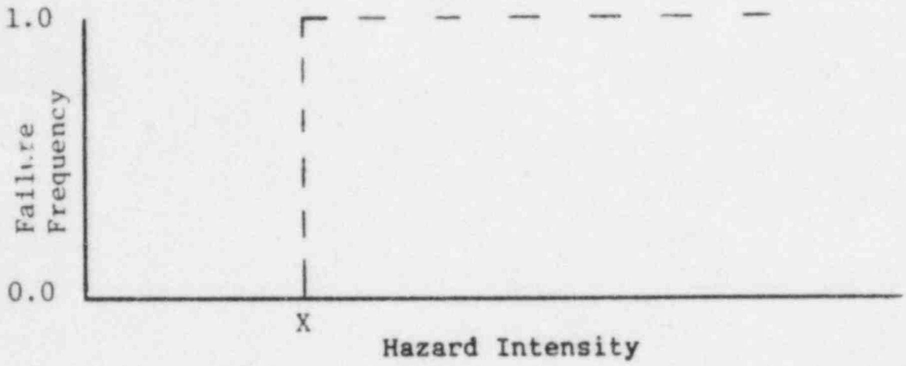
systems, and equipment. The methods used to perform this analysis and the consequent results are provided in Sections 5.0 and 6.0.

2.1.3 Fragility Analysis

As discussed in NUREG/CR-2300, the "fragility or vulnerability of a component is defined as the conditional frequency of its failure given a value of the response parameter".

As portrayed in Figure 2-1, this conditional failure frequency is typically represented by a discrete family of curves which display both randomness and uncertainty.

In this study, it was decided to represent the fragility curves by step functions as shown below.



The "point estimate" values of X were developed conservatively such that they are expected to be less than the mean value that would be determined from a complete propagation of the randomness and uncertainty parameters implicitly present in this type of evaluation.

Section 5.0 presents the development of fragility for this analysis.

2.1.4 Plant Systems and Event Sequence Analysis

A plant model was developed to identify potential initiating events (e.g., loss of off-site power and small LOCA) and model the systems available to mitigate these events (Section 3). This model development was based on work performed in the YNPS Probabilistic Safety Study (Reference 3) and follows guidelines discussed in NUREG/CR-2300.

First, potential plant initiating events were established by evaluating the response of plant structures and systems to the hazard. Second, initiating events impacting the plant response and mitigative systems in a similar manner were convolved into discrete categories. Third, mitigative system requirements w:re established for each initiating event category based on ensuring satisfaction of "critical safety functions", such as those listed below:

1. Reactivity Control

2. Main Coolant System (MCS) Inventory Control

3. MCS Pressure Control

4. Core Heat Removal

5. MCS Heat Removal

6. Containment Integrity

Event trees and fault trees were used to model plant response and were quantified as discussed below.

2.1.5 Release Frequency Analysis

The 1) hazard, 2) plant system and structure response, 3) fragility, and 4) plant systems and event sequence analyses were combined to establish the likelihood of the following basic parameters:

1. Core melt, including timing and conditions.

2. Core melt plus containment failure, including timing and other parameters required to assess off-site consequences.

Core Melt Frequency

The assessment of core melt frequency (Section 6.0) was based on quantification of the plant model (Section 3.0) using the hazard information from Section 4.0, fragility evaluations from Section 5.0, and non-hazard induced equipment failure frequencies developed in Sections 6.3 and 6.4.

Core Melt Characteristics

The timing and conditions of the potential core melt were based on the plant model results and the MARCH analyses performed in the YNPS PSS (Reference 3).

Release Frequency

The YNPS PSS results indicated that potential releases could be characterized into six discrete release categories. For example, release times varying from 1 hour to 50 hours and release energy rates from 500 Btu/hr to 80×10^6 Btu/hr.

From a review of the characteristics of these release categories, it was concluded that potential releases due to tornado and wind initiating events could be conservatively enveloped using one or more of these release categories. Section 7.0 explains the assignment of release categories to the spectrum of potential events resulting from a tornado/wind initiating event and the consequences of this release.

2.1.6 Consequence Analysis

The YNPS PSS included an assessment of potential effects for each of the release categories defined in that study. The analysis for the YNPS PSS was performed with the CRAC2 computer program and included plant-specific

radionuclide evaluations using the ORIGIN computer program and site-specific weather, topography, and evacuation information.

The major concern with direct use of this information involves the impact of tornado and wind events on evacuation and weather conditions. Both of these areas were, therefore, investigated. Section 7.0 discusses this investigation.

2.1.7 Risk Profile

The information developed by performing the analyses described in Sections 2.1.5 and 2.1.6 was combined to develop quantitative estimates of the following risk indices:

1. Individual acute fatality risk within 1 mile of the plant,

2. Societal latent cancer fatality risk per person within 50 miles of the plant, and

3. Person-rem exposure within 50 miles of the plant.

The first two risk indices are those developed in Reference 1 by NRC. The third risk index, person-rem exposure, is used in assessing the cost-benefit aspects of proposed plant design changes based on $1,000 per person-rem averted for the next 10 years. This is also based on guidance offered in Reference 1.

2.1.8 Changes in Plant Risk Profile

For each plant configuration, individual, societal, and person-rem risk levels were compared. This comparison provides a quantitative measure of the impact of each potential plant modification. Additionally, a comparison of the variation in the estimated core melt frequency was developed.

2.2 Comparison of Plant Risk to Safety Goals (Part 4)

Numerous proposals have been made in recent years to set numerical goals or guidelines to judge the acceptability or desirability of the

numerical risk levels calculated for nuclear power plants. Recently, the NRC has published preliminary safety goals and numerical design objectives (Reference 1). These goals and objectives can be stated quantitatively as follows:

Component of Risk	Quantitative Objective	Population Considered
Individual Acute Fatality Risk	5×10^{-7} per year	Within 1 mile
Societal Cancer Fatality Risk	2×10^{-6} per year per person	Within 50 miles
Cost-Benefit	\$1000 per person-rem averted	Within 50 miles
Large Scale Core Melt	10^{-4} per year	

Reference 1 also makes the following two key points:

1. "No further benefit-cost analysis should be made when it is judged that all of the design objectives have been met."

2. "The design objective for large-scale core melt is subordinate to the principal design objectives limiting individual and societal risks."

This information was used in assessing the cost-benefit relationship of backfits proposed to reduce the risk due to tornado and wind events and is discussed next.

2.3 Cost-Benefit Analysis (Part 5)

For the different plant configurations, all of which meet the individual and societal risk goals of Reference 1, an assessment of the justifiable costs to reduce these risk levels consistent with each upgrade considered was performed. As discussed in Section 2.1.8, this was based on \$1000 per person-rem averted per Reference 1 and taken over a 10-year period.

Next, these justifiable costs were compared to the actual cost required for the backfit. If the actual costs exceeded the justifiable costs, the backfit is not warranted.

An important aspect of this cost-benefit analysis is that it exceeds the explicit need for a cost benefit if "all of the design objectives have been met" as described in Reference 1. However, Yankee concluded that it was reasonable to apply the $1000 per person-rem averted to each backfit option to assess its cost-benefit characteristics.

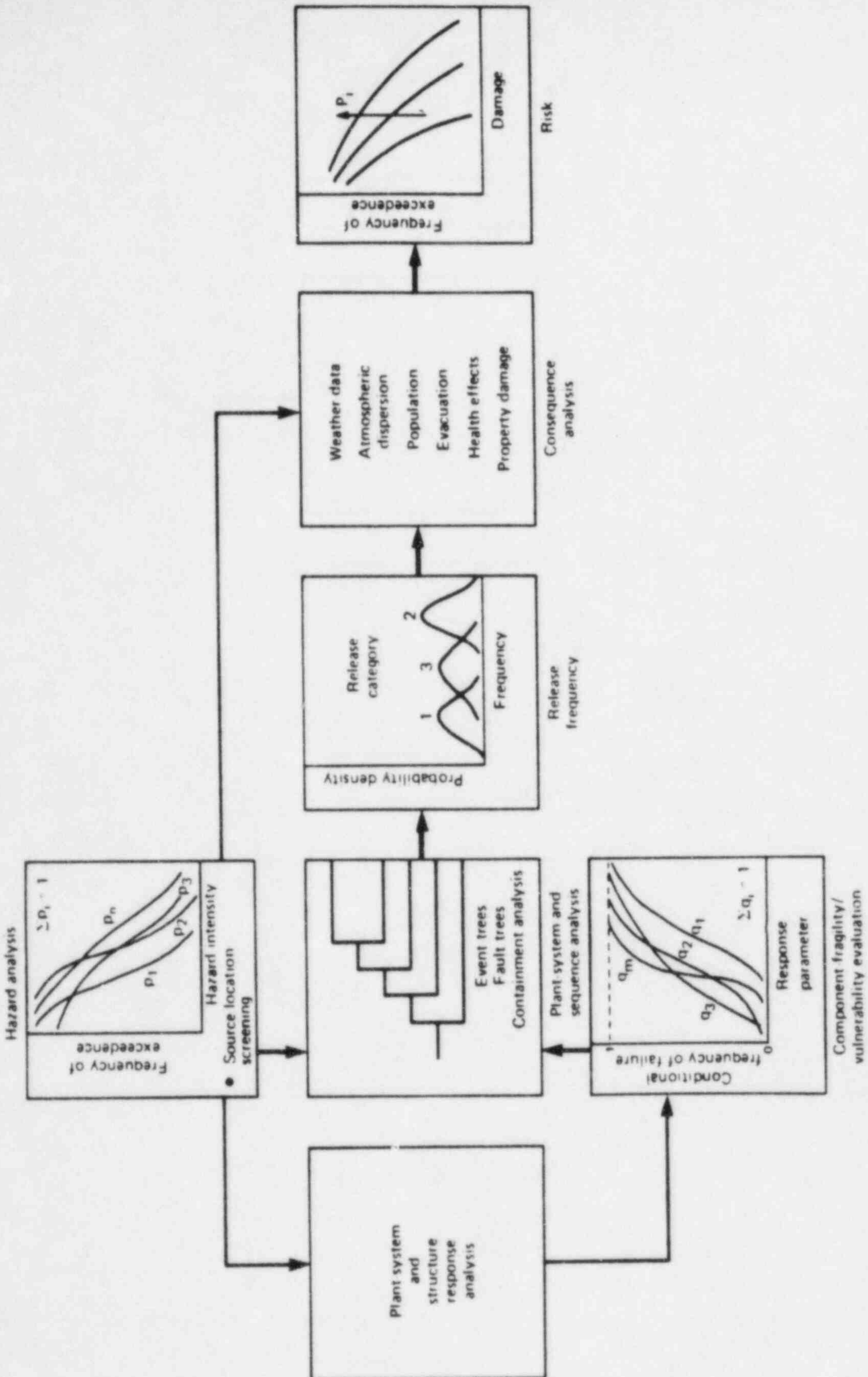


FIGURE 2-1

Risk Assessment Procedure for External Events

3.0 PLANT MODEL DEVELOPMENT

As discussed in Section 2.1.4, a plant model was developed to represent potential initiating events and the systems available to mitigate these events. This model development was based on work performed in the YNPS Probabilistic Safety Study, but was specialized to account for the unique characteristics of this assessment.

Each of the major steps involved in this development is discussed below. Section 6 develops and quantifies the model's logic expressions.

3.1 Initiating Event Definition

Sections 3.0 and 5.0 of the YNPS PSS provide detailed discussions of the approach taken to identify and quantify "random" accident initiators. A brief review is provided below because this information serves as the bases for the initiating event types quantified in this tornado/wind analysis.

- A Master Logic Diagram (MLD) was developed to serve as a road map for searching for accident initiators. In essence, the approach is based on a deductive evaluation of the plant design and operation and the potential perturbations of basic plant performance parameters that could impact continued plant operation.

  The outcome of this deductive process is a listing of twenty-seven categories of possible initiating event types which could impact plant operation. Table 3-1 provides a listing of these categories. Figures 3-1 and 3-2 display the MLD development.

- EPRI-NP-801 (Reference 4) was reviewed to determine if other initiating event categories existed that were not identified in the deductive development of the MLD.

- Personnel familiar with the plant design, operation, and transient performance characteristics reviewed the MLD categories and EPRI-NP-801 events. Any events not included in the MLD or EPRI-NP-801 were added.

- o Having completed the search for accident initiators, the impact on plant response of each initiator was evaluated to determine those initiators that had similar effects on the plant. From this review process, it was possible to convolve the possible initiators identified into nineteen specific initiating event categories. These nineteen events are listed in Table 3-2.

For the purposes of this analysis, the nineteen events can be further grouped into the following three basic categories:

1. Loss of Off-Site Power

   a. Plant trip

   b. Loss of ac

   c. Decrease in feedwater flow

   d. Decrease in steam flow

      1) Loss of vacuum

      2) NRV closure

      3) Turbine trip

   e. Degradation of dc power supply other than Bus No. 1

   f. Decrease in component cooling capability

   g. Decrease in service water delivery

   h. Decrease in control air delivery

2. Excessive Cooldown

   a. Excessive cooldown

b. Steam line break

c. Loss of dc Bus No. 1

3. Loss of Main Coolant System Inventory

a. Any LOCA

b. Reactor vessel rupture

c. Non-isolable LOCA outside containment

The bases for this categorization are provided below.

Category 1 (Loss of Off-Site Power)

The ten specific initiating event types included in this category can be treated as a loss of ac event because of their impact on plant response. For example, a loss of ac event envelopes plant trip, decrease in feedwater flow, decrease in steam flow, and decrease in either component cooling, service water, or control air events. The degradation of dc power supply event is taken into account because it is one of the support systems challenged in the logic model developed to represent those mitigative systems available to respond to a loss of ac.

Category 2 (Excessive Cooldown)

The three specific initiating events, excessive cooldown (a steam or feedline rupture affecting a single steam generator), steam line break (a break in a main steam line affecting all four steam generators), and loss of dc Bus No. 1 (impacts turbine trip and non-return valve closure), were treated together because they all result in an excessive plant cooldown.

An excessive plant cooldown event could lead to pressurized thermal shock concerns if not adequately controlled and mitigated. The characteristics of each of these three events are sufficiently comparable

relative to mitigative system performance that they could be treated conservatively by the same basic logic model.

Category 3 (Loss of Main Coolant System Inventory)

All loss of Main Coolant System inventory events can be treated as 1 event for 2 reasons.

1. Their likelihood is low, and

2. Mitigative system performance requirements can be defined to conservatively envelop a spectrum of break sizes.

3.2 Critical Safety Functions/Mitigative Systems

The YNPS PSS provides detailed discussions of the event sequence development for the three categories of initiating events provided above. In summary, they were based on a logical representation of those systems needed to ensure satisfaction of the following basic "critical safety functions":

1. Reactivity Control

2. MCS Inventory Control

3. MCS Pressure Control

4. Core Heat Removal

5. MCS Heat Removal

6. Containment Integrity

Figure 3-3 provides a generalized critical safety function-based event tree.

Each of the six critical safety functions (CSFs) listed above was reviewed with respect to wind/tornado events (off-site power loss assumed).

Mitigative systems were identified for further evaluation including location dependence.

3.2.1 Reactivity Control

The reactivity control CSF consists of two major categories: "insertion of negative reactivity" and "control of positive reactivity addition".

1. Insertion of Negative Reactivity

   Negative reactivity is inserted by the control rods following actuation of the Reactor Trip System. This consists of three basic functions:

   a. Detection of Need to Trip

      The need to trip is detected from various parameters monitored by the scram system. The perturbation of any of these parameters results in a reactor scram signal powered from dc Bus No. 1 being sent to the scram breakers (BK-1 and BK-2).

   b. Trip

      Trip initiation is also supplied from dc Bus No. 1 to open BK-1 and/or BK-2. DC Bus No. 2 supplies tripping power to BK-1 and BK-2.

      Failure of dc Bus No. 2 results in rod insertion because the rod drive holding coils are energized by this bus. A failure of dc Bus No. 1 results in tripping the scram breakers due to "fail safe" actuation of the main coolant flow under current/over current trip system.

c. Rod Insertion

Rod insertion is assured since the rods fall in by gravity. The integrity of the rod drive system is not challenged for hazard intensities not affecting containment integrity Chemical injection is not considered since the objective is to maintain the plant in a hot standby condition.

2. Control of Positive Reactivity Addition

To prevent the insertion of excessive positive reactivity following trip, Main Coolant System temperature must be controlled; therefore, control of steam removal and feedwater addition is required, as well as sufficient instrumentation.

a. Control of Steam Removal

Control of steam removal involves two phases:

- o Termination of normal steam removal through the turbine, and
- o "Post-trip" steam removal control.

1) Termination of Normal Steam Removal Through Turbine

Closure of turbine throttle and control valves terminates steam flow to the turbine. Support systems include tripping by dc Bus No. 1 and sensing by generator electrical relaying, reactor scram, or manual trip. Mechanical overspeed trip is the frontline backup to this tripping mechanism for a loss of load event. Further backup is provided by closure of the non-return valves (NRVs) in each of the four main steam lines.

2) "Post-Trip" Steam Removal Control

Turbine bypass would be unavailable because of loss of circulating water and loss of control air resulting from the loss of ac. The atmospheric steam dump system and steam generator code safety valves are available.

The atmospheric steam dump is sufficient to remove decay heat. These valves can be controlled remotely by the operator from the Control Room or locally by manual operation. Remote operation requires electrical power as follows:

- o Emergency 480 V ac Bus No. 1 (2 valves), and
- o Emergency 480 V ac Bus No. 3 (2 valves).

The size of these valves is such that a rapid cooldown would not occur if a valve failed fully opened. Furthermore, they can be manually isolated using upstream isolation valves.

Steam generator safety valves back up atmospheric steam dump. If atmospheric steam dump is not available, heat can be removed by allowing the Main Coolant System and consequently the steam generator fluid to heat up to the point of simmering the steam generator safety valves. If a valve should stick open, it can be closed (gagged) by the operator locally and challenged valves are of low capacity relative to the potential for an excessive cooldown.

b. Control of Feedwater Addition

Normal main feedwater would be unavailable due to the loss of off-site ac. There are five systems capable of supplying water to the secondary system. For each system, the operator has the ability to manually control the flow rate either by trimming

the number of running pumps or throttling the flow or recirculation from the pump. These systems and their vital auxiliaries are:

1) Electric Emergency Feedwater (EFW)

    a) 2400 V Bus No. 2 or 3

    b) Tanks TK-39 or TK-1

    c) DC Bus No. 3 or 1

    d) Flow path

        o Main feedwater

        o Blowdown

2) Steam Emergency Feedwater

    a) Main steam

    b) Tanks TK-1 or TK-39

    c) Flow path

        o Main feedwater

        o Blowdown via charging path or electric EFW path

3) Charging Pumps (three needed for success for short term; one pump needed after 1 day)

    a) 480 V Bus 4-1 through MCC-4, Bus 2 (MCC = Motor Control Center)

b) 480 V Bus 6-3 through MCC-2

c) 480 V Bus 5-2 through MCC-4, Bus 1

d) TK-39 or Safety Injection Tank (SIT)

e) Flow path

- o Charging to main feedwater
- o Charging to blowdown

4) ECCS Pumps

a) Emergency 480 V Bus 1, 2, or 3

b) Safety Injection Tank

c) Respective pumping train for energized 480 V Bus

d) Flow path

- o Blowdown header
- o Charging header to main feedwater

5) Safe Shutdown System (SSS)

a) Separate diesel and pump

b) Fire water storage tank

c) Flow path to blowdown

c. Instrumentation

In order to control heat removal, the operator must have instrumentation to detect the cooling conditions of the Main Coolant System. The instruments are supplied from Vital Buses No. 1 and 2 which are supplied from the station dc Buses No. 1 and 2 on the loss of ac. Other detection means include local readings at the vapor container penetrations, local installed instruments, and Safe Shutdown System instruments.

3.2.2 Main Coolant System Inventory Control

To control Main Coolant System (MCS) inventory, the operator must be able to detect the level or the core cooling effectiveness (if level is below the pressurizer), isolate the MCS, maintain this isolation, and make up for losses from the system.

1. Isolation

The Containment Isolation System protects the MCS from inventory loss. It is supplied from control air and the station 125 V dc battery buses. Inside containment, the pressurizer safety and relief valves, as well as the MCS boundary, must be intact.

o Pressurizer Power-Operated Relief Valve (PORV)

The pressurizer power-operated relief valve, if challenged, must reclose or the operator must take action to close the PORV block valve or treat the open PORV as a loss-of-coolant accident.

o MCS Safety Valves

The MCS safety valves are not expected to be challenged; however, if they were challenged and failed to reclose, a LOCA is assumed.

o MCS Pressure Boundary Integrity

The Main Coolant System boundary is not expected to be challenged by the spectrum of winds for which the containment is not challenged (i.e., $<$250 mph).

2. Makeup

Given containment isolation with the maximum allowable leak rate of 1 gpm, makeup would not be required to keep the core covered with water for a period of days. Additionally, the Safe Shutdown System has the ability to charge to the Main Coolant System to make up for normal leakage.

3. Detection

Level and temperature instrumentation power is supplied from Vital Buses No. 1 and 2 which, on a loss of ac, are supplied from dc Buses No. 1 and 2, respectively. The SSS provides additional detection and monitoring instrumentation.

3.2.3 Main Coolant System Pressure Control

This critical safety function is aimed at ensuring MCS integrity. It can be considered a subset of the CSF "MCS Inventory Control". The most important element of MCS pressure control is maintenance of MCS pressure below a threshold value at which the structural integrity of the MCS is threatened. Too low an MCS pressure is not critical because even saturated conditions will result in adequate core cooling as long as the core is covered and decay heat is being removed from the MCS. Core heat removal and MCS heat removal critical safety functions address this area.

For post-trip (i.e., decay heat power level conditions), the pressurizer PORV and code safety valves provide this protection. In fact, even these systems are not required if the MCS is being adequately cooled by feedwater addition and steam removal. Since failure to supply sufficient feedwater or remove steam is addressed by the MCS heat removal critical safety

function review, MCS pressure control is redundant and can be neglected with the following exception.

If an excessive cooldown of the MCS were to occur, it is possible that operator actions would be required to control the MCS pressure to preclude pressurized thermal shock (PTS) failure of the vessel. The events that could lead to this situation are characterized by steam or feedline ruptures plus mitigative system failures outside the current design bases of the plant. It is necessary, however, to address these events in this analysis because sequences are not limited to those within design bases. To address events that could lead to a PTS concern, a separate initiating event category was established, excessive cooldown. PTS type sequences resulting from other events such as those addressed by the "Loss of Off-Site Power" initiating event category are not important because of 1) the capacity of the steam removal and feedwater addition systems, and 2) control of these systems is specifically addressed in examining satisfaction of the MCS heat removal critical safety function.

3.2.4 Core Heat Removal

To maintain the core cooled, the Main Coolant System must remain intact to the steam generators and not be blocked. Additionally, enough water must be present to cover the core.

These conditions are verified by detection of Main Coolant System inventory control and core exit temperatures.

3.2.5 Main Coolant System Heat Removal

Heat removal from the Main Coolant System is credited only by secondary heat removal in this analysis. No credit is taken for primary feed and bleed cooling mechanisms. Steam removal is provided by atmospheric steam dump or steam generator code safety valves. Feedwater addition is provided as discussed under Reactivity Control (Section 3.2.1).

3.3 Event Sequence Analysis

Having identified the critical safety functions and systems which provide those functions, event trees can be developed for each of the three initiating event categories. Event sequences are discussed briefly here and in detail in Section 6.0.

3.3.1 Loss of Off-Site Power

The loss of off-site power category is the primary initiating event category for this analysis since plant history indicates that for most major storms in the vicinity of YNPS, the off-site power grid is affected. (Note, however, that only one complete loss of off-site power event has occurred in 24 years, its duration was 30 minutes and it was not weather-induced.) Furthermore, because of the hazards effect on off-site power, a complete loss of off-site power is conservatively assumed for all event sequences in this analysis.

Figure 3-4 provides the loss of off-site power event tree.

3.3.2 Excessive Cooldown

In reviewing the mitigative features required for this category, the same functions are required as for the loss of off-site power category with the addition of the non-return valves. As will be discussed in Section 6.0, this category is a negligible contributor to wind/tornado-induced core melt.

3.3.3 Loss of Main Coolant System Inventory

This category can be considered to consist of three event types:

1. Main Coolant System (MCS) pressure boundary violation involving piping or component physical failure.

2. Failure to isolate MCS bleed paths, such as the letdown portion of the Chemical and Volume Control System, and

3. Failure of MCS pressure relieving devices (i.e., PORV, safety valves).

As Section 6.0 discusses in detail, the first two LOCA event types above are negligible contributors to core melt for this analysis; the third is represented by the Relief Valve Challenges Event Tree (Figure 3-5). Figure 3-6 represents possible event sequences should a LOCA occur.

Event sequence logic expressions are developed and quantified in Section 6.0.

3.4 Identification of Critical Areas

Based on the systems required to maintain the critical safety functions identified above, the critical plant areas required for safe shutdown in a high wind event can be determined. The routing of each system or auxiliary was reviewed for locations.

3.4.1 Systems/Auxiliaries Vs. Critical Areas

1. DC Bus No. 1	Switchgear Room
2. DC Bus No. 2	Switchgear Room
3. Emergency 480 V AC	
1. Diesel Generator	Individual Diesel Cubicle
2. E480 V AC Bus	Safety Injection Building (SI Building) SI Building North Wall
3. Respective DC Buses for Breaker Control	1 & 2 - Switchgear Room SI Building North Wall 3 - SI Building

4. Emergency Motor Control Center (EMCC)	EMCC-1 - Switchgear Room EMCC-2 - SI Building EMCC-3, 4, 5 and 6 - Remote Shutdown Facility
5. Fuel Oil Storage	Southeast Yard

4. 480 V AC

1. Associated Emergency 480 V AC Bus	See Above
2. Cable from E480 V AC Bus through Manhole No. 3 to Switchgear Room	SI Building, North Wall
3. 480 V AC Bus	Switchgear Room

5. 2400 V AC Bus 2 or 3

1. E480 V AC Bus	See Above
2. 480 V AC Bus	See Above
3. Station Service Transformer Nos. 5 and 6	Switchgear Room
4. Nos. 2 and 3 Station Service Transformer Bus	Station Service Transformer Yard Pump Room
5. 2400 V AC Bus	Switchgear Room

6. DC Bus No. 3A

1. DC Bus No. 3	SI Building

2. Cabling DC Bus No. 3 through Manhole to 3A Bus	SI Building, North Wall
3. DC Bus No. 3A	Switchgear Room
7. TK-39, Primary Water Storage Tank	Southeast Yard
8. TK-1, Demineralized Water Storage Tank	Under VC, Outside Auxiliary Boiler Room (ABR)
9. Electric Emergency Feedwater Pump (Elect. Pump)	Lower Level Primary Auxiliary Building (LL PAB)
10. Steam EBFP	Auxiliary Boiler Room North Wall
11. MCC-4	LL PAB
12. MCC-2	Turbine Building Pump Room
13. ECCS Pump	SI Building
14. Vital Bus	Switchgear Room
15. Atmospheric Steam Dump	Non-Return Valve (NRV) Enclosure
16. SG Safety Valves	NRV Enclosure
17. Steam EBF to Main Feed Path	Auxiliary Boiler Room Pump Room Under VC
18. Elec. EBF to Main Feed	LL PAB Under VC Pump Room

19. Elec. EBF to Blowdown	LL PAB UL PAB Northwest Wall UL PAB North Wall Upper Pipe Chase
20. Charging Cross Connect to Main Feed	LL PAB Under VC Pump Room
21. Charging to Blowdown	LL PAB Upper Level (UL) PAB North Wall Upper Pipe Chase
22. SI to Blowdown	SI Building UL PAB Upper Pipe Chase
23. Charging Pumps	PAB Cubicle Area
24. Elec. EBF Supply Piping	LL PAB
25. Steam EBF Supply Piping	Auxiliary Boiler Room South Wall
26. Charging Supply Piping	LL PAB
27. SI Tank	Southwest Yard
28. SI Supply Piping	LL PAB SI Building
29. Safe Shutdown System	South Yard
30. Fire Water Storage Tank	South Yard
31. Safe Shutdown System Feed Conn.	UL PAB North Wall Upper Pipe Chase

32. Non-Return Valves	Switchgear Room Turbine Building (TB) West Staircase NRV Enclosure
33. Vital Instrument Detectors	Vapor Container Cable Tray House Pump Room LL PAB
34. Turbine Throttle Valves	Switchgear Room Turbine Building Mezzanine
35. Vital Bus	Switchgear Room

3.4.2 Location and Description of Critical Areas

From the list above, a table of critical areas was developed identifying system dependencies by area (Table 3-3).

Description of areas:

1. Auxiliary Boiler Room South Wall

   The supply piping from TK-1 passes through the Auxiliary Boiler Room south wall. If this wall were to topple, the supply piping could be faulted from TK-1 or TK-39 to the steam-driven emergency boiler feed pump.

2. Switchgear Room

   The 480 V and 2400 V electrical supplies for charging and electric-driven emergency feedwater come from the Switchgear Room. Additionally, the dc power to operate the switchgear, detection, and NRV actuation comes from the Switchgear Room.

3. Diesel Generator Cubicle

The diesel generators and their cooling air and fuel oil are located in their respective cubicles.

4. Safety Injection Building

The Safety Injection Building is critical for operation of the safety injection pumps and Battery No. 3. Isolated areas are critical for operation of electric-driven equipment such as electric-driven emergency feedwater and charging. The emergency 480 V ac bus is located in an area that is relatively well protected even if a structural cladding failure were to occur on the SI Building. The bus is protected on two sides by the inside wall of the diesel cubicle and the remote shutdown facility. On the remaining two sides it is protected by distance and the intervening SI pumps and piping.

5. SI Building North Wall

The north wall is important to maintaining continuity of emergency 480 V ac electric power to the Switchgear Room and 125 V dc to the diesel generators and emergency switchgear.

6. Station Service Transformer Yard

In order to supply stepped-up 480 V ac to the 2400 V bus, the station service transformer bus must not be faulted outside the building at the voltage regulators and transformers following an event. The structure supporting the station service transformer supply from off-site power may topple. If it should, an electrician could remove a portion of bus bar to disconnect the faulted station service transformer within a few hours. This action is not proceduralized.

7. Fuel Oil Tank

Needed for extended running of diesel generators.

8. TK-39, Primary Water Storage Tank

Needed as a primary water source to charging and electric-driven emergency feedwater and an alternate source to the steam-driven emergency feedwater.

9. Under Vapor Container

The electric emergency feed and charging to main feed path pass under the vapor container as do the main feed lines.

10. Lower Level Primary Auxiliary Building

The electric supply for Nos. 1 and 3 charging pumps comes through motor control center No. 4 in the lower level PAB. The electric-driven emergency feedwater pumps are located in the lower level PAB. The emergency feedwater cross-connect for electric and charging feed as well as the vapor container recirc line are located in the overhead and pass through the east end of the north wall.

11. Auxiliary Boiler Room North Wall

The steam-driven emergency feedwater pump is located on the north wall of the Auxiliary Boiler Room and the discharge pipe passes through it. This is an internal wall between the Auxiliary Boiler Room and Turbine Building Pump Room.

12. Pump Room

The Turbine Building Pump Room is the common junction of the main feedwater header with the steam emergency feed, electric emergency feed, and charging emergency feed headers. The power supply to charging pump No. 2 at motor control center No. 2 is in this area.

13. Upper Level Primary Auxiliary Building North Wall

The common junction of the blowdown feed path and the safety injection feed, charging feed, electric emergency feed, and Safe Shutdown System feed paths is located in this area.

14. Upper Level Primary Auxiliary Building West Wall

The electric emergency feed path and safety injection feed path are located in this area. The safety injection line is not expected to be faulted by any structural cladding failure in this area due to its size and wall thickness. The safety injection piping is expected to protect the electric emergency feed piping from damage due to cladding failure.

15. Upper Pipe Chase (Non-Radioactive Pipe Tunnel)

The individual blowdown lines pass through this area to reach the containment.

16. Primary Auxiliary Building Cubicle Area

The charging pumps and emergency feed line from charging are located in this area. This area is constructed of reinforced concrete.

17. Safety Injection Tank

The safety injection tank is the primary water source for the ECCS and the alternate supply to charging.

18. South Yard

The Safe Shutdown System and the fire water storage tank are located in the yard south of the vapor container.

19. TK-1, Demineralized Water Tank

The demineralized water storage tank is located outside the south wall of the Auxiliary Boiler Room and provides the primary water source for the steam-driven emergency feed pump and alternate source for the electric-driven emergency feed pump and charging.

20. Non-Return Valve Platform

The NRV platform is the one area for steam removal from the system and provides isolation in the event of a steam line rupture.

21. Turbine Building

The turbine throttle valves are the primary steam line isolation along with the main steam dump. A fault on the line to these valves can be isolated by the non-return valves.

22. Turbine Building West Staircase

Cabling for operation of vital equipment, such as the NRVs and atmospheric steam dumps, passes through this area. Additionally, this area provides access for the operator to other plant equipment following a severe event.

23. Cable Tray House

Signals to and from the vapor container pass through this area for detection and control.

24. Vapor Container

The vapor container provides containment of the atmosphere surrounding the Main Coolant System boundary and vital detection equipment.

3.5 Containment Assessment

3.5.1 Background

The design of the YNPS containment is such that it is cooled entirely by passive means. Heat transfer between the containment environment through the containment steel skin to the outside atmosphere is sufficient to maintain the containment temperature and pressure below design conditions for all design basis events.

The combination of structural, MARCH, CORCON-MOD1, and manual calculations performed in the YNPS PSS also showed that even core melt conditions should not result in containment pressures exceeding the ultimate pressure capacity of this structure, about 100 psia. This assessment included an evaluation of piping, valving, and electrical penetrations.

The same analysis performed in the YNPS PSS indicated that containment ultimate pressure responses were not extremely sensitive to the specific sequence that led to core melt and vessel failure.

3.5.2 Approach

In this tornado/wind study, there are additional considerations to be addressed because of the potential effects of the hazard on containment structural integrity.

Extremely high intensity events, equivalent wind velocities exceeding about 250 mph, have the potential to fail the containment directly. These events have frequencies less than 10^{-6}/yr even using the 95% confidence level hazard curve.

Because of this direct impact on containment integrity, and consequently core cooling capability, the YNPS PSS results could not be used directly. Instead, containment response was treated discretely as follows:

1. For hazard intensity levels not affecting containment integrity, YNPS PSS results were used; and

2. For hazard intensity levels resulting in containment failure, a core melt and early 1-hour direct release to the environment were assumed.

Sections 6.0 and 7.0 discuss this approach in more detail.

TABLE 3-1

Master Logic Diagram Initiating Event Categories

Increase in Main Coolant System Pressure

Decrease in Main Coolant System Pressure

Reactor Vessel Rupture

Steam Generator Tube Rupture

Very Small LOCA

Small LOCA

Intermediate LOCA

Large LOCA

Increase in Main Coolant System Inventory

Dilution

Rod Withdrawal

Rod Ejection

Inadvertent Rod Insertion

Rod Drop

Boration

Increase in Main Coolant System Flow

Decrease in Main Coolant System Flow

Increase in Steam Flow

Feedwater Induced Increase in Secondary Heat Removal

Decrease in Steam Flow

Feedwater Induced Decrease in Secondary Heat Removal

Degradation of the AC Power Supply

Degradation of the DC Power Supply

Decrease in Component Cooling Water Delivery

Decrease in Service Water Delivery

Decrease in Control Air Delivery

Non-Isolable LOCA Outside the Containment

TABLE 3-2

Initiating Events

Excessive Cooldown

Steam Line Break

Very Small LOCA

Small LOCA

Intermediate LOCA

Large LOCA

Steam Generator Tube Rupture

Plant Trip

Loss of AC

Decrease in Feedwater Flow

Loss of Vacuum

NRV Closure

Turbine Trip

Degradation of DC Power Supply

- Single Bus
- Double Bus
- All Buses

Loss of Service Water

Loss of Component Cooling Water

Loss of Control Air

Non-Isolable LOCA Outside Containment

Reactor Vessel Rupture

TABLE 3-3

Critical Areas Vs. Systems

Pumping System	Elec. EBF		Steam EBF			SSS	Charging System		ECCS		
Flow Path	MW FW	Blowdown	MW FW	EEBF Blowdown	Charge Blowdown	Blowdown	MW FW	Blowdown	Blowdown	Blowdown Charge - MW FW	Remarks
ABR South Wall			X	X	X						Flow path to steam EBF suction.
Switchgear Room	X	X					X	X			Emergency AC to normal AC.
DG Cubicle	X	X					X	X	X	X	Emergency AC supply may affect emergency bus cabling from batteries 1 and 2 and emergency bus.
SI Building	X	X					X	X	X	X	
SIB North Wall	X	X					X	X	X	X	2400 V bus supply may be faulted by fault in SST yard.
SST Yard	X	X									
Fuel Oil Tank	X	X					X	X	X	X	Emergency DG long-term primary supply.
TK-39	X	X					X	X			Cross-connect piping under VC.
Under VC	X			X	X		X			X	
LL PAB	X	X		X	X		X			X	Electric EBFP, MCC-4, ECCS suction.
ABR North Wall			X	X	X						Inside ABR at steam EBF MCC-2, overhead is junction of charging, steam, electric, and MFW.
Pump Room	X		X	X	X		X			X	
UL PAB North		X		X	X	X		X	X	X	ECCS, Elec. EBF and charging to blowdown.
UL PAB West		X		X					X	X	ECCS and Elec. EBF to blowdown
Upper Pipe Chase		X		X	X	X		X	X		Blowdown to containment charging pumps and headers.
PAB Cubicle							X	X		X	
SI Tank							X	X	X	X	ECCS supply and charging alternate
South Yard						X					Safe Shutdown System and fire water storage tank
TK-1			X	X	X						Demineralized water supply.
MRV	X	X	X	X	X	X	X	X	X	X	Steam removal and steam line isolation.
Cable Tray	Note 1										
Vapor Container	X	X	X	X	X	X	X	X	X	X	Access and MRV actuation.
TB West Staircase	X	X	X	X	X	X	X	X	X	X	
Turbine Building	Note 2										

Notes: 1 - The cable tray house is the preferred path for detection, the SSS provides a redundant path.
2 - The Turbine Building provides steam line isolation following trip. If the steam line fails, the MRVs provide backup.

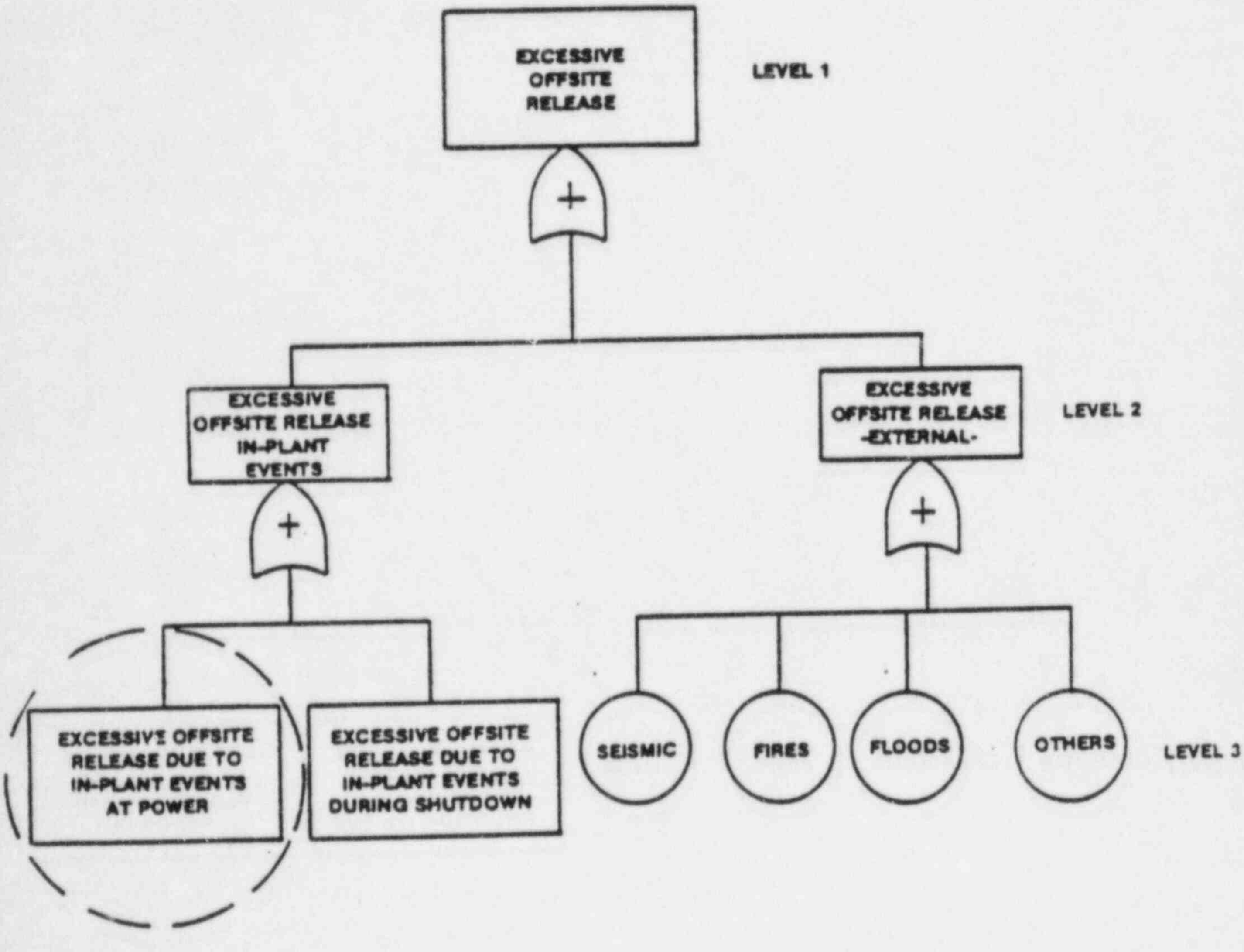


FIGURE 3-1

Master Logic Diagram - All Events - Excessive Release

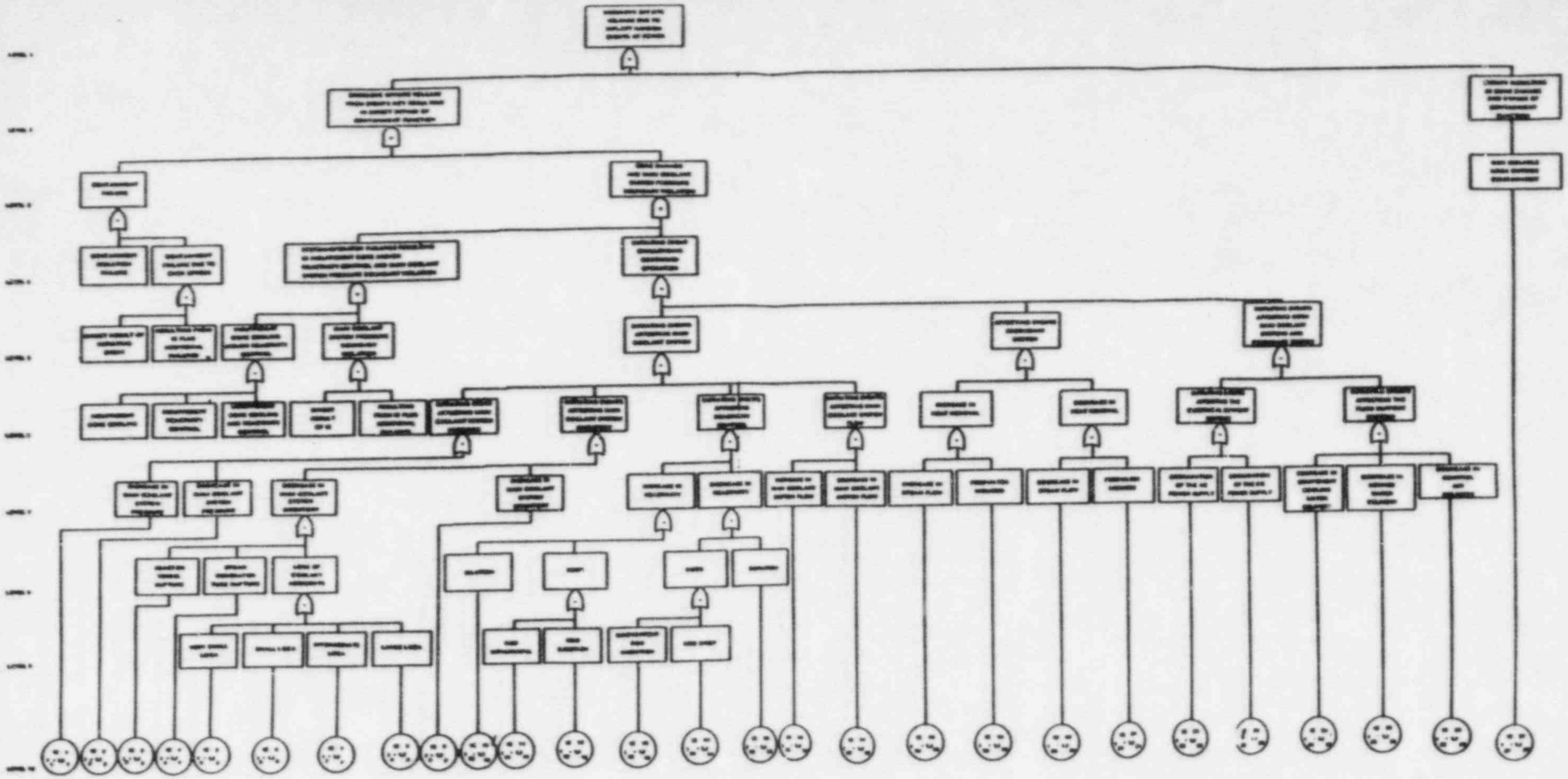


Figure 3-2

MASTER LOGIC DIAGRAM-EXCESSIVE
RELEASE DUE TO INPLANT EVENTS

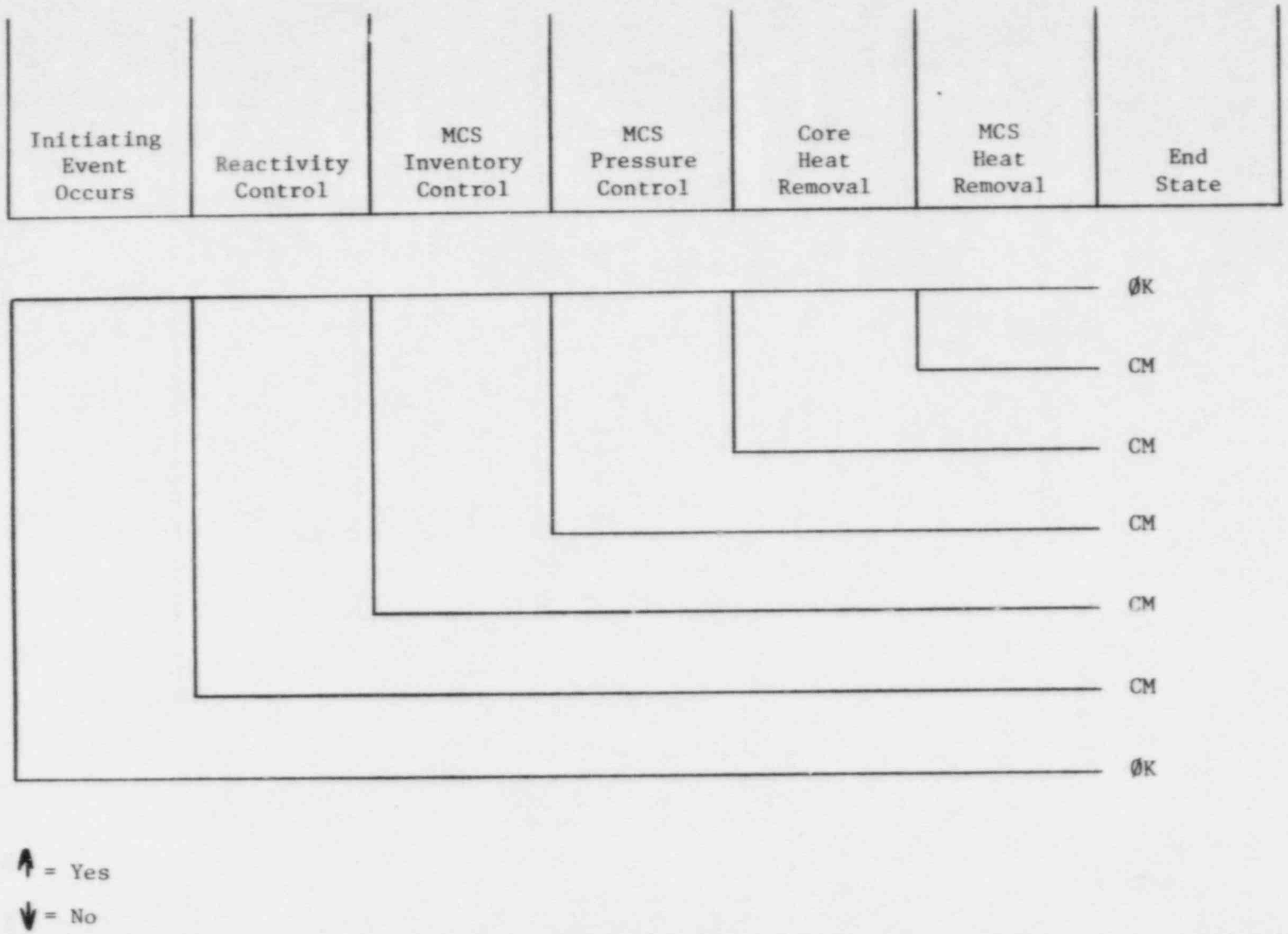


Figure 3-3

Generalized Critical Safety Function Based Event Tree

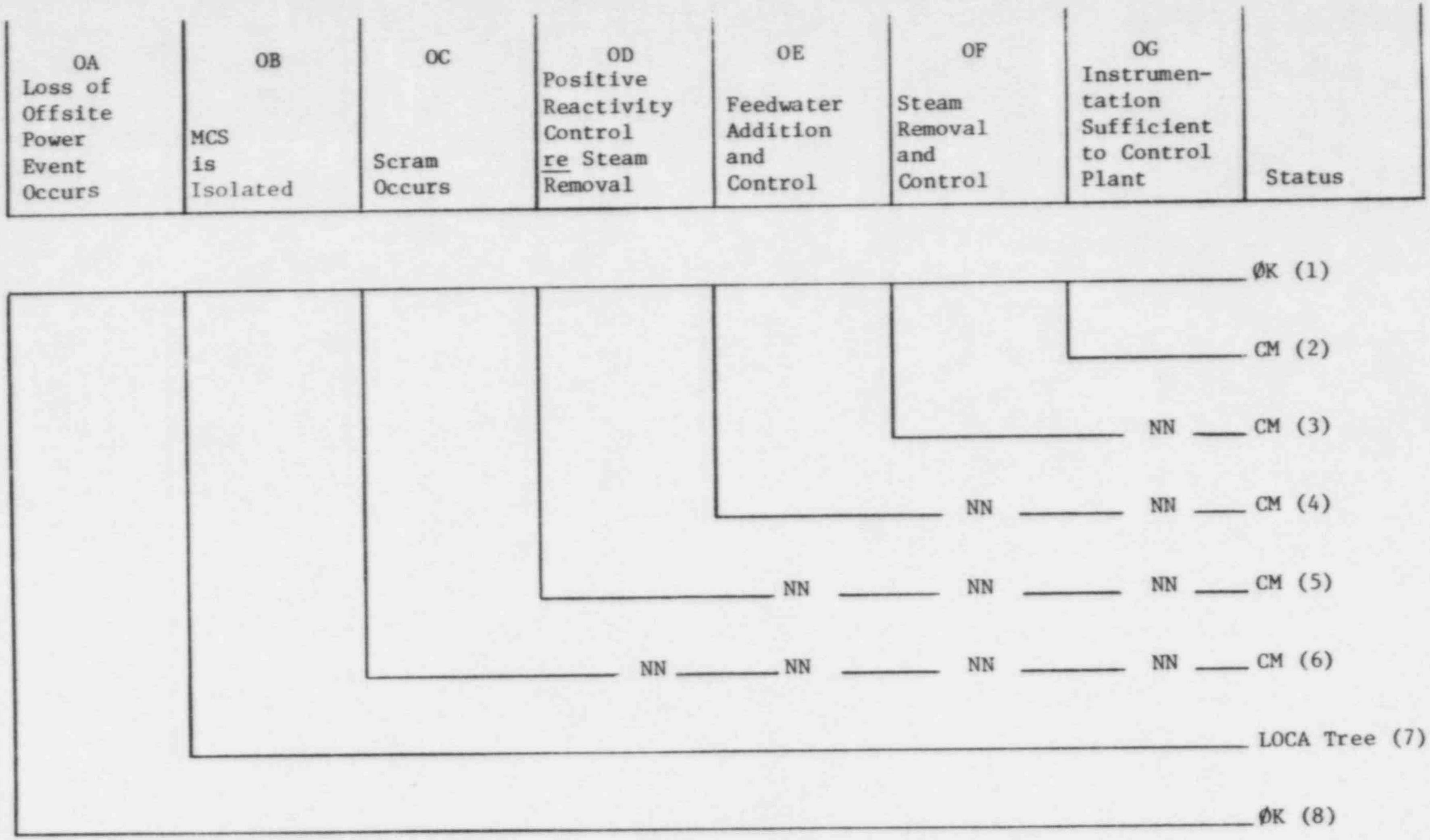
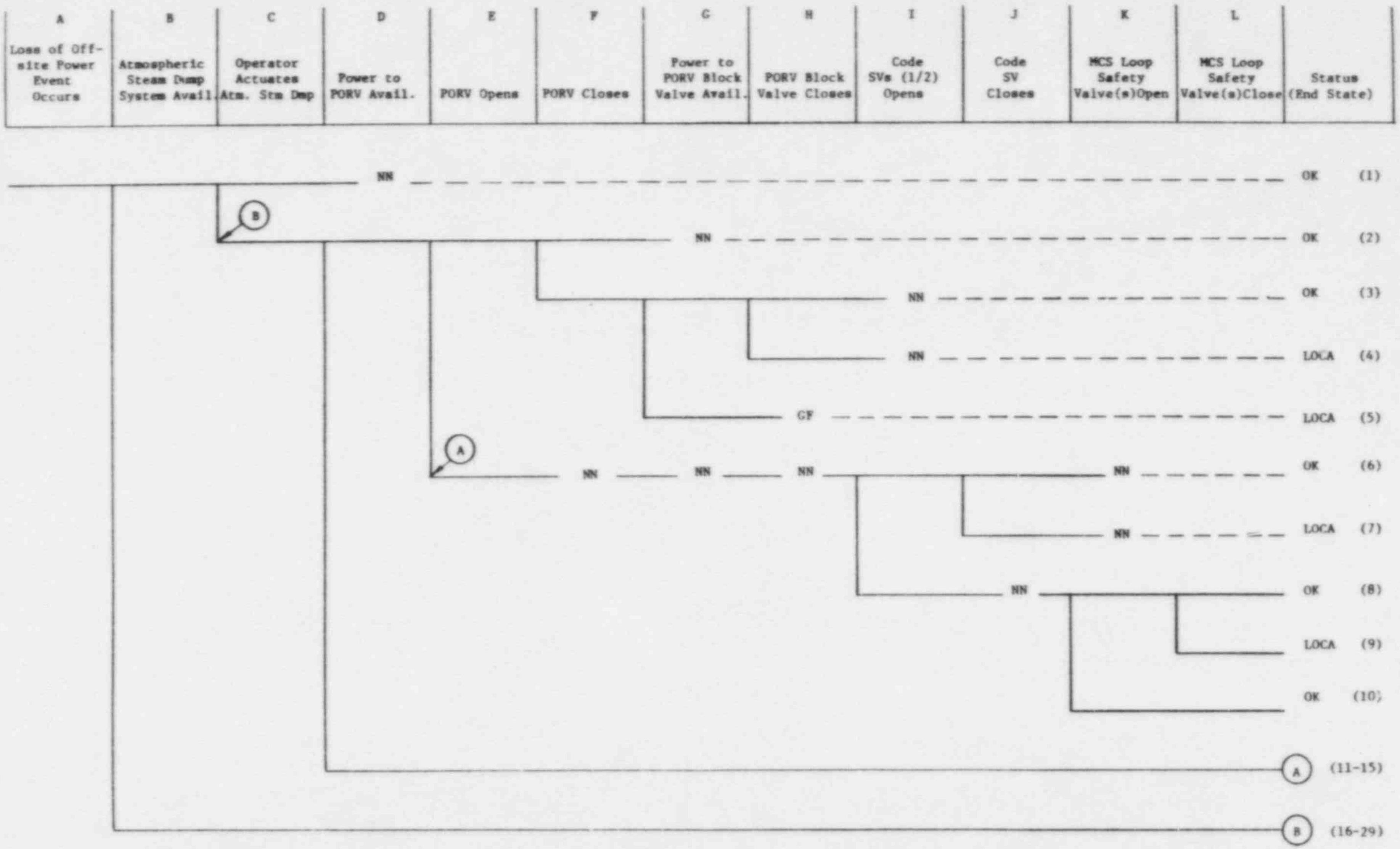


Figure 3-4

Loss of Offsite Power Event Tree

RELIEF VALVE CHALLENGES EVENT TREE



Legend NN : Not Necessary
GF : Guaranteed Failure

Figure 3-5

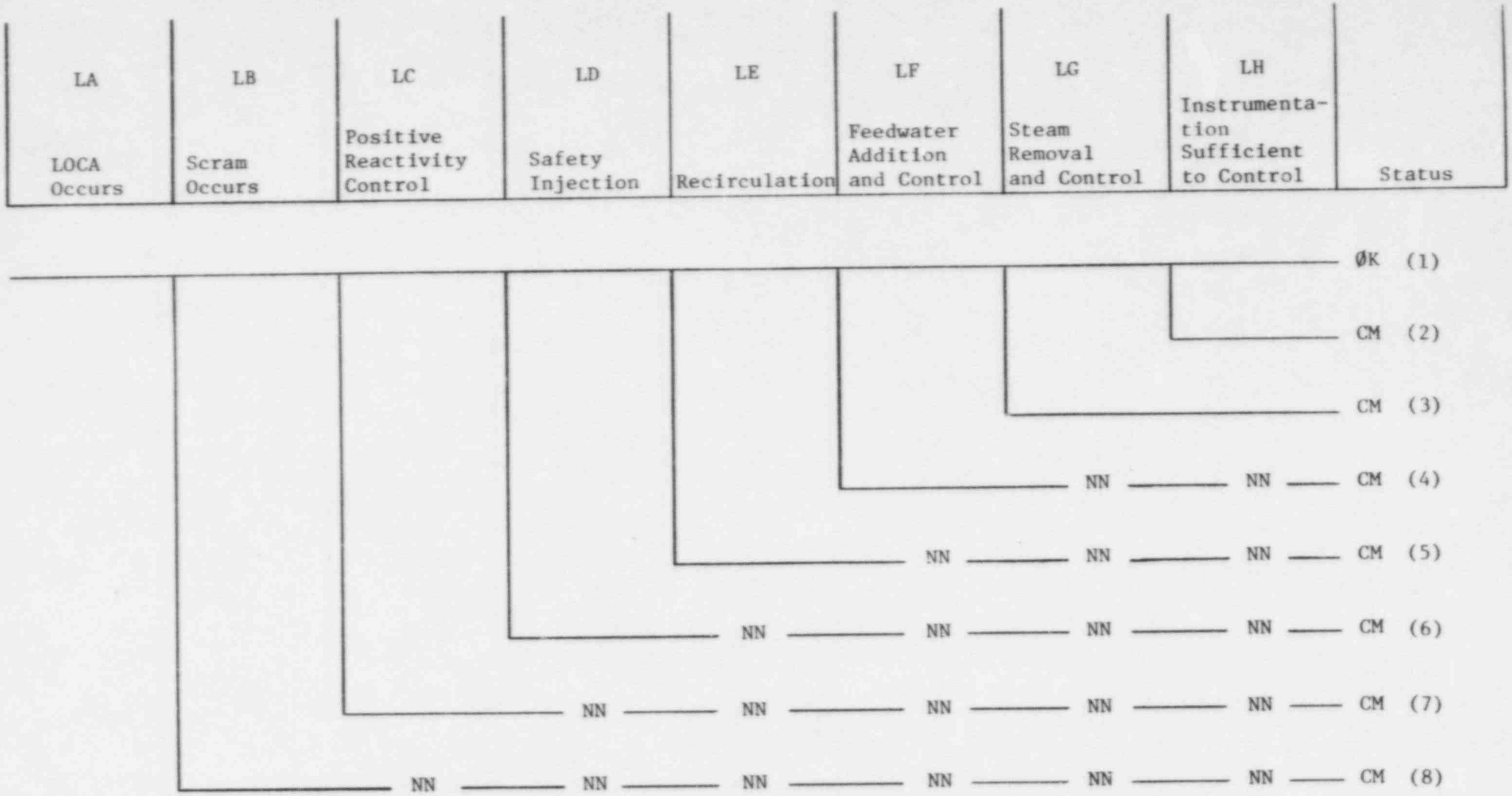


Figure 3-6

LOCA Event Tree

4.0 HAZARD INFORMATION

This section discusses the development of the wind/tornado hazard curves used in the cost-benefit analysis.

4.1 Wind and Tornado Hazard Probabilities

Straight wind and tornado probabilities associated with the expected value and upper 95% confidence intervals were determined for input into the cost-benefit analysis. The hazard probability estimates are the same as those identified in Reference 5 with the following adjustment. Upon review of the NRC's assessment in Appendix A of Reference 5, Yankee identified an incorrect input for the local region area in the tornado analysis. As correctly noted on page 5 of Reference 5, the local region area should be 28,945 square miles. However, the computer's printout listing in the Appendix A shows that a local region area of 20,560 square miles was used in the analysis.

The result of this incorrect area in the analysis is an overestimation of the annual probability of exceedance. Yankee has subsequently "corrected" the analysis in Reference 5 to account for the incorrect input area. The net effect on the results due to this correction is given in Table 4-1.

Using the "corrected" tornado hazard results in conjunction with the straight wind probabilities, given in Table 4-2, the straight wind/tornado hazard curves were developed. These hazard curves are given in Figure 4-1 for the expected and upper 95% confidence level.

TABLE 4-1

Tornado Hazard Probabilities

Annual Probability	Tornado Windspeeds, mph			
	NRC Expected Value	Corrected Expected	NRC Upper 95% Level	Corrected Upper 95% Level
10^{-4}	40	40	87	85
10^{-5}	122	110	174	165
10^{-6}	188	175	244	230
10^{-7}	248	235	291	285

TABLE 4-2

STRAIGHT WIND HAZARD PROBABILITIES

Annual Probability	Straight Windspeeds, mph Expected Value	Upper 95% Level
10^{-1}	57	61
10^{-2}	70	78
10^{-3}	82	94
10^{-4}	94	110
10^{-5}	107	127
10^{-6}	119	143

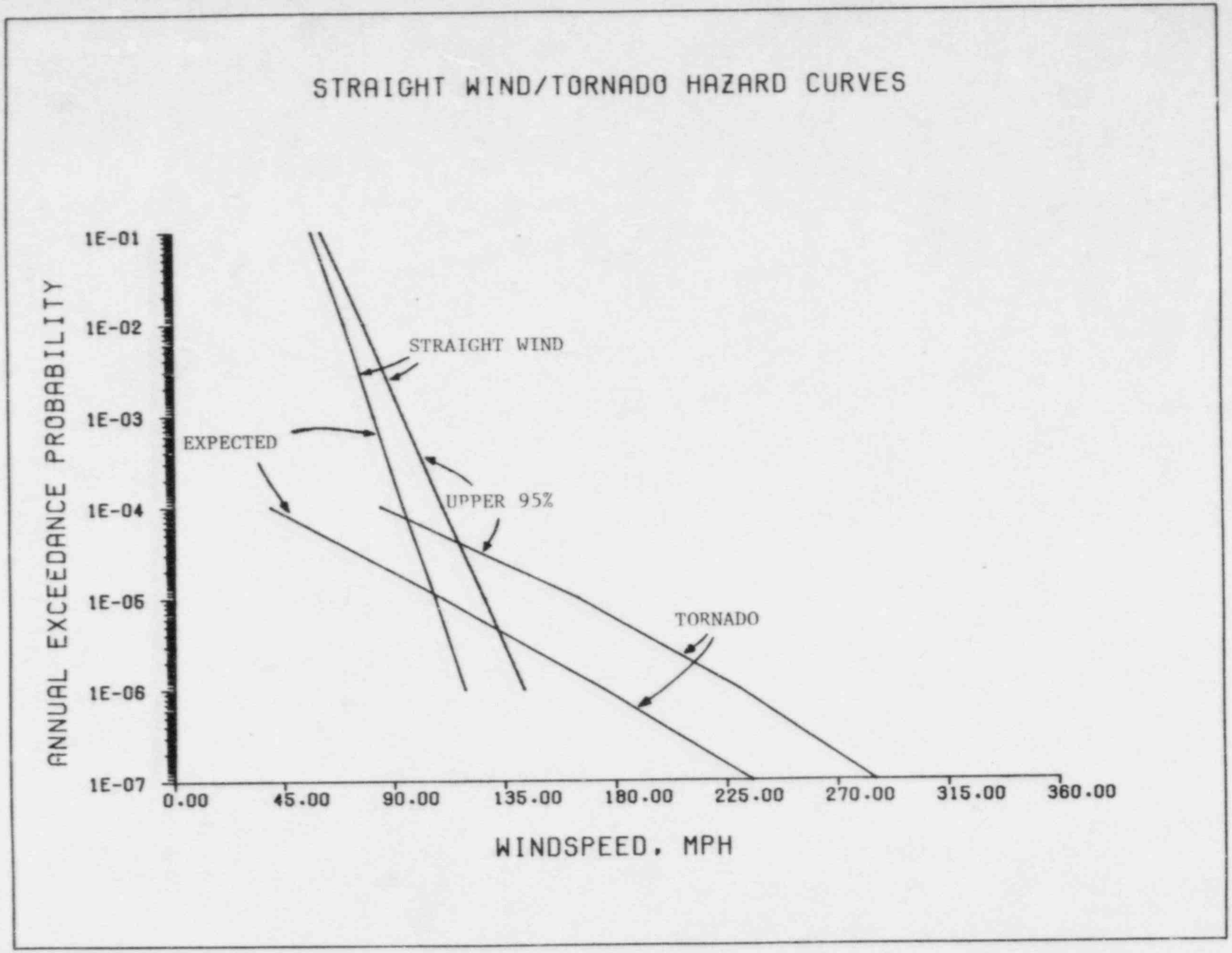


FIGURE 4-1

HAZARD CURVES

5.0 FRAGILITY ANALYSIS

Fragilities for critical structures and components are a basic input into the cost-benefit analysis. Critical structures and components necessary for hot safe shutdown are discussed in Sections 3.0 and 6.0. These structures/components are listed in Table 5-1. Fragilities for these structures/components are modeled as a step function with failure probability stepping from 0.0 to 1.0 at "failure" windspeed.

5.1 Failure Criteria

Failure criteria for structures/components evaluated for the cost-benefit analysis are as follows:

a. Block Walls

For unreinforced masonry block walls, failure is defined as exceeding the ultimate out-of-plane lateral pressure loading as governed by the tensile strength of the mortar at failure. Ultimate tension values of 22.7 psi (normal to bed joint) and 45.7 psi (parallel to bed joint) were used. These ultimate tension values are consistent with Reference 6 and are 1.62 and 1.11 times the design tension allowables in Reference 7 for normal and parallel values, respectively.

b. Tanks

Based on the respective tank drawings for TK-1, TK-28 and TK-39 a design wind load of 25 psf on the projected area of the cylindrical surface is specified. For the fire water tank, the tank specification calls for a design wind pressure equivalent to 18 psf. These wind design pressures were adjusted upward to ultimate lateral pressure loads on the projected area of the cylindrical surface. Ultimate lateral pressure loads at "failure" are 30 psf for TK-1, TK-28, and TK-39, and 35 psf for the fire water tank.

c. Equipment Enclosures

Ultimate lateral pressure loads at "failure" were determined by adjusting wind design pressures upward by the ratio of factor of safety to the original design allowable stress increase for wind loadings.

d. Roofs

Roof uplift was evaluated using the deadweight of the roofs with an allowance for fastening of the roof deck to the roof girders.

5.2 Structure/Component Windspeed Capacity

The conversion of ultimate lateral pressure loads to straight wind and tornado windspeeds is based on information presented in References 8 and 9. The methods in the references are actually design tools, which from a given straight or tornado wind, a design pressure is calculated. For this analysis, the methodologies were used to calculate an ultimate best-estimate windspeed for the structure/component equivalent to the calculated ultimate lateral pressure.

For the straight wind calculations, Exposure Category B, as defined in Reference 3 (wooded area), was used. Considering the location of the plant site with respect to the surrounding terrain, the topography and vegetation of the surrounding region, and the sinuosity of the river valley, it was determined that Exposure Category B best reflects plant site conditions. Buildings and structures were analyzed as main wind-force resisting systems or components and cladding, as appropriate. The straight windspeeds determined after Reference 8 are equivalent to fastest mile windspeeds, 33 feet above the ground for Exposure Category C and are directly compatible with the straight wind hazard curves.

For the tornado wind calculations, a technique from Reference 9 was used. The calculated tornado windspeeds are equivalent to peak tornado windspeed, rotational and translational, and are directly compatible with the tornado hazard curves.

For the tornado portion of the analysis, the conversion of ultimate lateral pressure load to tornado windspeed was based on the dynamic pressures of the tornado windspeeds. The atmospheric pressure drop loading was assumed as not producing the controlling load for structures/components in the cost-benefit analysis for the following reasons:

a. The calculated tornado pressure drop is based on maximum tornado windspeed. Structures/components with a finite horizontal exposure would experience an average tornado windspeed less than the maximum and, therefore the actual tornado pressure drop would be somewhat less.

b. A detailed analysis of the Battery Room demonstrated that the existing ventilation system of the room is capable of relieving the pressure drop differential to the point where the ΔP loading for the 10^{-5} upper 95%, 165 mph tornado is negligible.

c. For the four tanks, the atmospheric pressure drop loading is quite small compared to the hydrostatic loading when the tanks are full; in addition, the tanks are vented.

d. Most of the structures/components analyzed are not airtight and, given that the tornado pressure drop is developed over 5 to 6 seconds for the tornados of interest, adequate time is available for venting before any appreciable pressure differential could build up.

e. The seismic SSS structures/components will be adequate for the 10^{-4} ΔP loading.

Based on the above discussion, it was concluded that the controlling tornado loading is the dynamic wind pressure and that the atmospheric pressure drop loading for existing structures/components is negligible. Note the conceptual design for the SSS Pump House calls for an airtight structure, as such, this structure will be designed for load combinations including the atmospheric pressure drop.

Table 5-2 lists the structures/components analyzed, their ultimate lateral pressure loads at failure, and wind capacities for both straight and tornado winds. The calculated windspeeds were determined by application of the techniques presented in References 8 and 9.

TABLE 5-1

Structures/Components for Fragility Evaluation

Auxiliary Boiler Room South Wall T1J2

Auxiliary Boiler Room Interior Wall T1G2

Lower Level Primary Auxiliary Building Wall P1E1

Lower Level Primary Auxiliary Building Wall P1E2

Upper Level Primary Auxiliary Building Wall P2F1

Upper Level Primary Auxiliary Building Wall P2F2

Safety Injection Building South Wall D1Z1

Safety Injection Building South Wall D1Z2

Safety Injection Building West Wall D11051

Safety Injection Building West Wall D11052

Diesel Generator Building West Wall D11053

Diesel Generator Building North Wall D1V1

Diesel Generator Building North Wall D1V2

Diesel Generator Building North Wall D1V3

Safety Injection Building North Wall D1X1

Demineralized Water Storage Tank, TK-1

Safety Injection Tank, TK-28

Primary Water Storage Tank, TK-39

Non-Radioactive Pipe Tunnel

Battery Room Wall T292

Battery Room Wall T2G3

Battery Room Wall T2G4

Turbine Building SW Stairwell Wall T1H2

Turbine Building SW Stairwell Wall T1121

Primary Auxiliary Building Roof

Non-Return Valve Enclosure

Fire Water Tank

Fire Pump House Enclosure

Safe Shutdown System Pump House

Cable Spreading Room Walls

Cable Spreading Room Roof

TABLE 5-2

Wind Capacities of Structures/Components

Structure/Component	Ultimate Lateral Pressure Load, psf	Wind Capacities, mph	
		Straight Wind	Tornado
Walls			
T1J2	22.8	122	162
T1G2 (Interior Wall)	23.1	122	162
P1E1	54.2*	197	222
P1E2	54.2*	204	222
P2F1	38.2*	165	186
P2F2	38.2*	171	186
D1Z1	9.7	103	93
D1Z2	9.7	103	93
D11051	10.4	107	81
D11052	7.7	92	70
D11053	7.5	91	69
D1V1	26.9	172	156
D1V2	26.9	172	156
D1V3	26.9	172	156
D1X1	16.3	134	121
T292 (Interior Wall)	44.4*	161	186
T2G3 (Interior Wall)	44.4*	145	186
T2G4 (Interior Wall)	44.4*	170	186
T1H2 (Interior Wall)	27.3*	126	145
T1121 (Interior Wall)	30.5*	120	154
TK-1	30.0	191	161
TK-28 (SIT)	30.0	179	164
TK-39	30.0	179	164
Non-Radioactive Pipe Tunnel	50.6*	173	176
Primary Auxiliary Building Roof	25.0	121	129
Non-Return Valve Enclosure	26.5*	135	119
Fire Tank	35.0*	190	182
Fire Pump House Enclosure	26.8*	156	119
Safe Shutdown System Pump House**	60.0*	258	178
Cable Spreading Room Walls	8.0	69	65
Cable Spreading Room Roof	10.6	73	80
Cable Spreading Room W/Fix	64.7***	196	186

*This calculated value assumes a "seismic upgrade" to YCS has been installed.

**To be constructed in 1985.

***This calculated value assumes structure upgraded to design for 10^{-4} event.

Note: Wind capacities for interior walls are conservative since these walls are not subject to direct wind pressure loading.

6.0 QUANTIFICATION OF PLANT MODEL

The purpose of this section is to discuss the release frequency analysis. This involves development and quantification of logic expressions representing the plant model (Section 3.0) to conservatively determine the likelihood of:

- o Core melt, including timing and conditions.

- o Core melt plus containment failure, including timing and other parameters required to assess off-site consequences.

The process involved in assessing each of these parameters is discussed below.

6.1 General Discussion of Core Melt Frequency Analysis

As discussed in Section 3.0, the core melt frequency analysis consists of 1) identification of possible sequences leading to core melt, and 2) quantification of these sequences. Initiating events were categorized into three specific types: 1) loss of off-site power, 2) excessive cooldown and 3) loss-of-coolant accidents. Mitigative system requirements for each event type were established based on maintenance of the "critical safety functions" listed below:

1. Reactivity Control

2. MCS Inventory Control

3. MCS Pressure Control

4. Core Heat Removal

5. MCS Heat Removal

6. Containment Integrity

Section 3.0 described the systems available to support maintenance of these critical safety functions and their locations at the plant. As discussed in Section 3.0, event trees and fault trees were used to translate this information into a plant model.

The purpose of this section is to translate the qualitative plant model into logic expressions and quantitative results.

6.2 Initiating Event Frequency

The frequency of each of the three initiating event types is dependent on the hazard intensity. This relationship is discussed below for each event type.

6.2.1 Loss of Off-Site Power

To assess the likelihood of a loss of off-site power due to weather related effects, two distinct sources of information can be used, as follows:

1. Historical data at the YNPS site and other plant sites, and

2. Convolution of the hazard curves with plant switchyard and transmission line capacities.

Sections 6.2.1.1 and 6.2.1.2 discuss the available historical data. The second information source is discussed in Section 6.2.1.3. Evaluation of this information also considered the following two important questions while assessing the likelihood of a loss of off-site power:

1. Does the hazard event resulting in a loss of off-site power also impact the structural integrity of the plant?

2. What is the function relating the frequency of off-site power loss with the duration of loss?

6.2.1.1 YNPS Data

The YNPS has operated for about 24 years. During this period, there has been only one complete loss of off-site power. This occurred on November 9, 1965 during the "Great Northeast Blackout". The plant was in cold shutdown at the time. A nearby hydro unit was used to re-energize the plant in about 30 minutes.

The loss of off-site power was not caused by local weather conditions. Thus, the evidence is 0 loss of off-site power events due to local weather conditions in 24 years at the YNPS.

6.2.1.2 Industry Data

6.2.1.2.1 Data

The following evaluation of weather-related loss of off-site power plant trips is based on the EPRI/NSAC data in Appendix A of draft report, "Losses of Off-Site Power at U.S. Nuclear Power Plants", dated May 1984.

Weather-Related (1) Plant Trips From LOSP

Cause	LOSP <1/2 HR	LOSP >1/2 HR	Total	LOSP >3 HRS	LOSP >6 HRS	LOSP >9 HRS
Storm(2)	5	3	8	0	0	0
Storm/Salt(3)	3	2	5	2	1	0
Lightning(4)	7	4	11	1	1	0
Tornado	2	1	3	1	0	0
Total	17	10	27	4	2	0

Notes: (1) Weather-related is used because some events resulted from a combination of weather and other failures.

(2) Storm includes wind, snow, ice or some combination thereof.

(3) Events that resulted from salt on the switchyard insulators were separated since they were not applicable causes for inland sites. These events occurred at Millstone and Pilgrim.

(4) Storms may be associated with lightning.

The total number of site years in the EPRI data is 533.

The frequency of LOSP vs. time using the maximum likelihood estimate is tabulated below.

	T > 0	T > 1/2 HR	T > 3 HRS	T > 6 HRS	T > 9 HRS
Total	.051	.019	.008	.004	.002*
Storm & Tornado**	.021	.008	.002	.002*	.002*
Tornado	.006	.002	.002	.002*	.002*

* 1/533 assumed vs. 0.

** Salt causes and lightning causes excluded.

Note that the overall industry frequency for a loss of off-site power event due to storms and tornados exceeding 3 hours is less than 2×10^{-3} per year. Additionally, the total frequency of a loss of off-site power caused by storms and tornados is about a factor of 10 higher at 2.1×10^{-2} per year. There were no events in which off-site power losses exceeded 6 hours due to tornados and storms. (The longest loss was 4 hours as discussed below.)

6.2.1.2.2 Discussion of Data

Tornados

The three events involving tornado-induced loss of off-site power occurred at the following plants:

1. Dresden, 11/12/65, tornado passing north of station resulted in a loss of all transmission facilities. One 138 kV line was restored in 4 hours.

2. Browns Ferry, 4/3/74, four of five 500 kV lines and one of two 161 kV lines failed. Several transmission line towers were down. Off-site power remained available from the remaining 161 kV line.

3. Arkansas Unit 1, 4/7/80, multiple line losses occurred, but the backup line remained available.

Thus, although there have been three plant trips caused by local tornado activity, only one event caused a complete loss of off-site power. And, for this event, off-site power was restored within 4 hours. Additionally, none of these events involved a direct hit at the plant. The evidence is therefore zero direct hits in 533 plant years of operation, and one event in 533 years of plant operation resulting in a complete loss of off-site power for 4 hours.

Storms

Excluding salt and lightning related events, there have been eight events caused by storms. Of these eight events, there have been only three

events in which off-site power was unavailable for greater than 30 minutes; no events exceeded a 3-hour loss. As assessed for tornados, none of these events resulted in structural failures at the affected plant.

6.2.1.3 Switchyard and Transmission Line Capacity

In addition to historical data, a convolution of hazard curves with switchyard and transmission line capacities can provide useful information regarding loss of off-site power likelihood. The on-site switchyard structure at YNPS is designed for a 70 mph straight wind. The off-site transmission lines are designed for a 100 mph straight wind, limited by the capacity of the supporting poles.

The plant's two independent off-site power supplies - Cabot and Harriman lines - would be available up to about 100 mph, their design rating. On-site switchyard failure (at about 70 mph) would result in a plant trip and would prevent powering the plant with the Harriman line. Power from the Cabot line would most likely remain available after switchyard failure since the station service transformer is tapped off upstream of the switchyard.

From Section 4.0, the exceedance frequencies of these events are provided below:

Hazard Intensity (mph)	Exceedance Frequency Per Year			
	50% Hazard Confidence		95% Hazard Confidence	
	Tornados	Winds	Tornados	Winds
70	4×10^{-5}	1×10^{-2}	2×10^{-4}	3×10^{-2}
100	2×10^{-5}	4×10^{-5}	7×10^{-5}	5×10^{-4}

Without crediting the inherent design conservatism in the switchyard and transmission line capacities, the frequency of a plant trip due to wind-induced switchyard failure would be about 1×10^{-2} per year for straight winds and about 4×10^{-5} per year for tornados. The frequency of off-site power loss resulting from severe transmission line failure would be about 4×10^{-5} per year due to winds and about 2×10^{-5} per year for tornados.

Note that the exceedance frequency versus hazard intensities used above for tornados are based on the YNPS site area. In fact, the two independent power supplies - Cabot and Harriman - cover distances of about 20 miles and 3 miles, respectively. Thus, the overall frequency of a tornado-induced loss of off-site power could be higher than stated above. This area dependence was investigated, as discussed below.

References 10 and 11 document investigations involving transmission line wind loadings performed by Dr. Lawrence A. Twisdale of Applied Research Associates, Inc. These papers were reviewed to assess the impact of transmission line length on the overall yearly frequency of failure due to wind and tornado loadings. The major conclusion relative to tornado-induced transmission line failure is quoted below from Reference 10.

> "For lines that exceed 10 miles in length, tornado winds may dominate the combined risk curve at fastest mile speeds as low as 70-80 mph."
>
> Additionally, "The relationship between transmission line probability and point target probability for this example is a factor equal to approximately 10 times the length of the line (mile)."

For the Cabot line, this relationship would yield a transmission line failure frequency due to tornados of about 4×10^{-3} per year and 1.4×10^{-2} per year, using the 50% and 95% hazard curves, respectively. For the Harriman line, the corresponding values would be 6×10^{-4} per year and 2.1×10^{-3} per year. Thus, the total frequency of losing a single line is 4.6×10^{-3} per year, 50% hazard, or 1.6×10^{-2} per year, 95% hazard, due to tornados.

There is insufficient plant specific data to validate these findings since the plant has only operated for about 24 years, with no tornado-induced single line losses. The industry data discussed in Section 6.2.1.2 demonstrated a frequency of 6×10^{-3} per year for a loss of off-site power due to tornados. However, for a complete loss of off-site power to occur at YNPS, either multiple - at least two - lines must be affected or the switchyard and Cabot line must be affected. Additionally, the longest (industry-wide, tornado-induced) loss reported was 4 hours, indicating that major structural damage did not occur.

Though a plant trip would occur on loss of a single line at YNPS, the conditional frequency of core melt is extremely low unless the plant itself incurs damage. If either line is intact, all normal plus emergency plant equipment can be powered. In addition to the individual frequency of Harriman or Cabot line loss due to tornados, a more important question is their dual failure.

Dual line loss - Cabot and Harriman - could occur if the following events occurred:

1. Tornado damage at the plant affecting the switchyard and Cabot line, where Harriman and Cabot join;

2. A tornado fails either the Cabot line or Harriman line, and the unaffected line fails "randomly"; or

3. A tornado(s) fails both lines during a short period of time (less than a 1-day period).

The frequency of Case 1 was discussed previously as falling in the range of 4×10^{-5} per year to 2×10^{-4} per year (50% and 95% confidence). The frequency of Case 2 is negligible. And, the frequency of Case 3 is believed to be bounded by Case 1 because the Cabot and Harriman lines leave the plant in opposite directions. A "double" hit would be required to fail both of these lines unless the strike occurred at the switchyard.

For hazard intensities sufficient to fail the transmission poles, greater than 100 mph, restoration of off-site power to the plant through these lines would involve major repairs. From 1 to 4 days is estimated to be required to perform these repair actions. Another option available to restore off-site power to the plant is described below.

A nearby hydroelectric facility, Sherman Station, could be used to supply power to the plant. Cabling could be run over the ground, to the 2400V station service transformers at the YNPS. Depending on the damage to Sherman Station, it is estimated that this action could be completed within 2 to 4 days. Cabling availability is not anticipated to be a problem.

6.2.1.4 Combined Data

6.2.1.4.1 Tornados

The "generic" data developed by EPRI demonstrates a point estimate value of about 6×10^{-3} per year (3 events in 533 plant years) for plant trips caused by tornados. The frequency of a complete loss of off-site power is a factor of 3 lower at 2×10^{-3} per year (1 event in 533 plant years). For this event, the tornado did not "hit" the plant; off-site power was restored within 4 hours.

There has not been a tornado-induced loss of transmission lines connected to the YNPS. Based on switchyard and transmission line capacities and the hazard curves, long-term losses of off-site power frequencies were estimated to range from 2×10^{-5} per year, 50% hazard, to 7×10^{-5} per year, 95% hazard. The following table summarizes the available information.

Loss of Off-Site Power Duration (hours)	EPRI Data (yr^{-1})	Plant-Specific Capacity/ Hazard-Based (yr^{-1})	
		50% Hazard	95% Hazard
>0	2×10^{-3}	4×10^{-5}	2×10^{-4}
>4	0	-	-
>24	0	2×10^{-5}	7×10^{-5}

The frequency of a plant trip due to partial line losses ranged from 4.6×10^{-3} per year, 50% hazard, tc 1.6×10^{-2} per year, 95% hazard, as discussed in Section 6.2.1.3. EPRI data indicates 4×10^{-3} per year (2 events in 533 years).

It is difficult to confirm the accuracy of frequencies in the 10^{-4} to 10^{-5} per year range because these frequencies indicate extremely rare events. And, as stated earlier, there have been no direct hits at any plant.

Since this study is focused on assessing the cost-benefit relationship of backfits to increase the capacity of the plant to direct hits, the plant-specific information is most appropriate. This acknowledges that the total loss of off-site power frequency due to tornados is probably higher than those values in Columns 3 and 4, above. However, unless the tornado were a direct hit, the cost-benefit characteristics of additional plant hardening would be unaffected. Thus, the information in Columns 3 and 4 is appropriate.

A final question centers around short-term losses (i.e., less than 24 hours). This is important because random equipment failure frequencies are dependent on the period of time they must operate. Electrical equipment such as diesel generators and dc power are the most sensitive to off-site power loss duration. From the table above, however, there is little difference (a factor of 2 to 3) between the frequency of losses exceeding 0 hours and 24 hours. Thus, for this analysis, the frequency for losses exceeding 0 hours will be conservatively used.

6.2.1.4.2 <u>High Winds</u>

The generic data developed by EPRI demonstrates a point estimate value of about 1.5×10^{-2} per year (8 events in 533 years) for plant trips caused by storms. Of these 8 events, only 3 events, frequency of 5.6×10^{-3} per year, resulted in losses exceeding 30 minutes. No events resulted in losses exceeding 3 hours.

There have been no storm-induced complete losses of off-site power at the YNPS. Several plant trips have occurred due to single line losses, lightning strikes for example. Based on switchyard and transmission line capacities and the hazard curves, long-term losses were estimated to be 4×10^{-5} per year, 50% hazard, to 5×10^{-4} per year, 95% hazard. The following table summarizes the available information.

Loss of Off-Site Power Duration (hours)	EPRI Data (yr^{-1})	Plant-Specific Capacity/ Hazard-Based (yr^{-1})	
		50% Hazard	95% Hazard
>0	1.5×10^{-2}	1×10^{-2}	3×10^{-2}
>0.5	5.6×10^{-3}	-	-
>3	0	-	-
>24	0	4×10^{-5}	5×10^{-4}

Long-term losses are rare events. There have been none greater than 3 hours throughout the industry. Therefore, judgement is required in developing an exceedance frequency versus time function. It was judged that the following relationship is reasonable, probably conservative.

Loss of Off-Site Power Duration (hours)	Exceedance Frequency (50% Hazard)	Exceedance Frequency (95% Hazard)
>0	1×10^{-2}	3×10^{-2}
>0.5	4×10^{-3}	1×10^{-2}
>3	1×10^{-3}	3×10^{-3}
>24	1×10^{-4}	1×10^{-3}

The 0.5-hour point was determined by reducing the 0-hour point by a factor ($1.5 \times 10^{-2}/5.6 \times 10^{-3}$) derived from the EPRI data.

The 3-hour point was determined by reducing the 0-hour point by a factor of 10 consistent with the EPRI Generic Experience Data derived as for storm and tornado causes in Section 6.2.1.2.1. The 24-hour point is based on the transmission line capacity, rounded up to the nearest decade.

For this analysis, the frequency of losses exceeding 0.5 hours will be conservatively used.

6.2.1.5 Results

For the purposes of this analysis, the frequency (per year) of a complete loss of off-site power caused by a wind/tornado hazard will be conservatively taken as:

Hazard	Frequency Per Year 50% Hazard Confidence	95% Hazard Confidence
Wind	4×10^{-3}	1×10^{-2}
Tornado	4×10^{-5}	2×10^{-4}

These values are believed to be conservative for the following reasons:

- o Switchyard/transmission line failure is assumed to occur at design windspeeds.
- o The generic industry data is based on any complete loss of off-site power caused by wind/tornado events. This analysis considers off-site power loss coincident with plant damage which is less likely.
- o These frequencies represent power outage durations exceeding 0.5 hours for winds and 0.0 hours for tornados. Quantification of all event sequences in this analysis (Section 6.3) assume outage durations of 24 hours.
- o From the hazard curve (Section 4.0) these values correspond to hazard intensities in the range of 70 to 80 mph. Off-site power at YNPS (partial - the Cabot line) is likely to be available up to about 100 mph hazard intensity.

6.2.2 Excessive Cooldown

As discussed in Section 3, this event was reviewed separately because an excessive plant cooldown event could lead to pressurized thermal shock concerns if not adequately controlled and mitigated. The excessive cooldown event consists of the following three specific initiating events, which are sufficiently comparable relative to mitigative system performance that they can be treated conservatively by the same basic logic model. These events are:

1. Excessive cooldown (affecting one steam generator)

   a. Steam line break upstream of the non-return valves (NRVs),

   b. Feedline break downstream of the feedline check valves,

2. Steamline break downstream of the NRVs (affecting all four steam generators), and

3. Loss of dc Bus No 1.

For Events 1 and 2, above, the non-return valves must be capable of closing, and the feed line to the affected steam generator must be closed. (For Case 1, automatic closure of the NRVs is not required if they close properly as a check valve.) Operability of the non-return valve actuators requires the availability of any two of the three dc buses for successful automatic 2 out of 3 low steam line pressure trip, or, dc Bus 1 or 3 for manual trip from the Main Control Room. Feedline isolation can be initiated remotely by closing the feedwater regulating valve with control air or closing the feed header isolation valve by non-IE 480 V ac from Bus 6-3. The feedline isolation can be initiated locally by manually operating any of the following valves: feed header isolation valve, feedline stop valve, or feedwater regulating valve to the affected steam generator. The feed and steam removal capability is the same as that for the loss of off-site ac event.

For the third event, loss of dc Bus No. 1, automatic turbine trip is not available due to loss of the dc bus. Steam line isolation by the turbine throttle valves or non-return valves must be successful to prevent an excessive cooldown and possible pressurized thermal shock concerns.

Automatic NRV closure on low-low steam line pressure is reduced to a single 2 out of 2 logic train powered by dc Buses 2 and 3. Non-return valve manual actuation is available only from dc Bus No. 3.

It is reasonable to assume that if a failure of dc Bus No. 1 were to occur during the high wind event, the loss of off-site ac will also occur since the "wind capacity" of dc power is significantly higher than ac power. This would result in isolation of steam flow by turbine throttle valve closure since the loss of load to the turbine will cause the turbine to trip on mechanical overspeed, thus terminating the cooldown.

For Events 1 and 2, the initiating event, high winds or tornados, will not challenge the main steam and feed piping for windspeeds below that which may challenge the structural integrity of the containment at 250 mph. The likelihood of simultaneous random piping failure is extremely remote.

For a loss of dc Bus No. 1 to result in an uncontrolled excessive cooldown, both of the following failures must occur:

1. Failure of an additional dc bus or NRV equipment failure, and

2. Turbine overspeed trip failure.

Considering the capacity of the dc power supplies and NRV equipment as well as random failures, the overall yearly frequency is less than 10^{-7} per year. Thus, this event is a negligible contributor and will not be addressed further as a specific initiating event.

6.2.3 Loss-of-Coolant Accidents

This initiating event category can be considered to consist of three specific event types, as discussed in Section 3.3.3.

1. Main Coolant System (MCS) pressure boundary violation involving piping or component physical failure,

2. Failure to isolate bleed paths such as the letdown system, and

3. Failure of pressure relieving devices such as the pressurizer power-operated relief valve and pressurizer code safety valves.

The analysis performed to identify these events and their likelihood is discussed below.

6.2.3.1 Piping/Component Physical Failures

The majority of the MCS piping and components which make up the system pressure boundary are located within the Vapor Container (VC). The structural integrity of the VC has been demonstrated for windspeeds in excess of 250 mph. Therefore, for the windspeeds being considered here, a failure of the VC and MCS pressure boundary components within the VC need not be considered below a 250 mph windspeed.

Piping connected to the MCS that extends outside of the VC is routed through the radioactive pipe tunnel to the Valve Room area of the Primary Auxiliary Building (PAB). These areas are constructed of a minimum of 2 feet of reinforced concrete. This offers a level of protection greater than that of the VC. Therefore, a failure of this pressure boundary piping need not be considered for the windspeeds considered in this analysis.

6.2.3.2 Isolation Failures

The potential exists for pressure boundary failure if inadvertent actuation of the isolation valves that establish the pressure boundary were to occur. The valves used inside the VC are electrically motor-operated. The only failure mechanism that could result in their inadvertent actuation is in the control portion of the valve circuitry. This equipment is located in the Main Control Room (MCR) and Switchgear Room (SWGR) areas of the Turbine Building. The MCR is constructed of reinforced concrete. The SWGR is also constructed of reinforced concrete with the exception of the north wall, an internal wall to the Turbine Building. This wall is constructed of concrete blocks reinforced with additional columns for seismic considerations. This wall is capabl of withstanding a 10^{-8} wind or 7×10^{-7} per year tornado

at windspeeds of 170 mph and 186 mph, respectively, using the 50% hazard. Corresponding exceedance frequencies using the 95% hazard are $<10^{-8}$ for winds and 5×10^{-6} per year for tornados. Therefore, a tornado or wind-induced failure of this control circuitry resulting in an inadvertent actuation of a motor-operated valve is not anticipated due to wall failure.

Lines which run outside the VC to the Valve Room area of the PAB require isolation by operation of the Containment Isolation System (CIS). This insures that the primary pressure boundary remains intact and that all normally open lines connected to the MCS are isolated. These lines are investigated below.

Failure of the letdown system to isolate can lead to a challenge to the Main Coolant System Inventory Control and Pressure Control Critical Safety Functions.

On a loss of off-site power, the Control Air System is de-energized and control air system pressure drops to zero within about 3 to 4 minutes from the loss of the bus or a maximum of six minutes after the loss of ac, assuming a 2-minute coast down on 480 V Bus 4-1. The bleed line isolation valve, LC-V-222, is held open by control air. Therefore, the bleed line will close within a maximum of 6 minutes of the initiating event. Additionally, the operator is directed by the immediate actions of the loss of AC supply emergency procedure to trip (close) the bleed line isolation valve, LC-V-222. If the bleed line isolation valve failed to close, the operator can isolate the letdown by closing in line valves CH-MOV-525 or CH-MOV-527, if electrical power can be restored to non-IE 480 V Bus 5-2. The operator can isolate letdown manually by locally closing manual bleed line isolation valve CH-V-715 in the upper level Primary Auxiliary Building Valve Room (a room with a 2-foot-thick reinforced concrete structure).

The operator would have a time period measured in hours to perform this action because of the small capacity of the letdown system. This conditional initiating event frequency coupled with the hazard frequencies result in a negligible overall contribution to the frequency of a LOCA.

Other potential letdown paths such as the sample system were also reviewed. They were found to function the same or better than the letdown system. Since their probability of being open upon the event is much less than that of the letdown system, these paths are negligible contributors to the event.

6.2.3.3 Relief Valve Failures

The relief valves connected to the Main Coolant System are listed below:

- Pressurizer Power-Operated Relief Valve (PORV) (1)
- Pressurizer Code Safety Valves (2)
- Main Coolant System Loop Safety Valves (4)

PORV

The PORV has a setpoint of about 2350 psi and a throat ID of about .9 inches. Dependent on operator action and equipment availability to operate the atmospheric steam dump valves, it is possible that the MCS pressure could increase to 2350 psi following a loss of off-site power event.

Without turbine bypass, which is unavailable on a loss of off-site power, the secondary temperature rise is sufficient to result in a MCS temperature rise; consequent swelling of the MCS could lead to MCS pressure increases beyond the setpoint of the PORV, if the secondary is allowed to challenge the secondary code safety valves.

Secondary code safety valves have a setpoint about 175 psi higher than the turbine bypass. This equates to a secondary temperature at secondary safety valve setpoint about 23^{o}F hotter than at turbine bypass setpoint. Additionally, during a loss of off-site power, MCS hot leg temperatures will be 30 to 40^{o}F higher because of the reduction in MCS flow rates during natural circulation.

RETRAN-based analyses show that the operator has between 5 and 10 minutes to manually initiate the atmospheric steam dump system to preclude a challenge to the PORV.

Pressurizer Code Safety Valves

The two pressurizer code safety valves are set at approximately 2500 and 2550 psi, respectively. Each valve has an effective throat ID of about 1 inch.

These valves would not be challenged unless 1) the operator failed to actuate the atmospheric steam dump system (or it is unavailable), and 2) the PORV failed to open. If both of these events occurred, the first code safety valve could be challenged. Only if it failed to open, would the second high set valve be challenged.

Main Coolant System Loop Safety Valves

Each of the four MCS loops has a small - 100 gpm - liquid relief valve with a set pressure of about 2750 psi. The only possibility for these valves to be challenged is if the following scenario occurred:

1) Operator fails to actuate atmospheric steam dump system (or it is unavailable), and

2) The PORV fails to open, and

3) Both pressurizer code safety valves fail to open.

As discussed in Section 3.0, a simple event tree was developed to depict the sequences that could result in a LOCA due to relief valve challenges, Figure 3-3. This relief valve event tree, when quantified, represents the likelihood of a LOCA resulting from a wind/tornado-induced loss of off-site power. It provides the development of Top Event "A" (the initiating event) of the LOCA event tree, Figure 3-6.

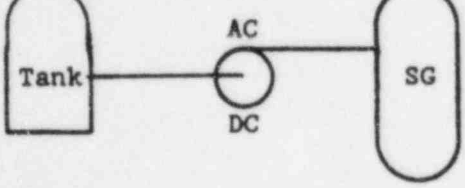
Quantification of the relief valve event, which yields the LOCA initiating event frequency, is discussed in Section 6.6.2.

6.3 Top Event Development

The approach taken to develop mitigative system failure frequencies accounted for both hazard induced and "random" failures. Random failure frequencies were based largely on results of the YNPS PSS. Failures induced by the tornado/wind hazard are failures caused by structural or other physical damage.

For each of the mitigative systems modeled, the physical locations containing equipment within these systems were identified (Section 3.0). Having identified potential random failures and potential location failures, logic expressions for mitigative system failure can be written. The following example illustrates the approach.

Assume a simple system consisting of an electric pump, control power, and flow path as follows:



Locations of each component follow:

Tank - isolated outside all buildings.

Piping from Tank to Pump - underground until entering Room A and joining the pump.

Pump - in Room A.

AC Power - supplied by Diesel Generator in Room B
and routed through Room C.

DC Power - supplied by Batteries in Room D
backed up by Charger (Room D) which is powered by the Diesel
Generator through Room E.

Discharge Piping - Through rooms with CAPACITY exceeding 250 mph after
leaving Room A.

Reviewing the above, a failure expression for the system can be written.

Tank is outside so the only location hazard failure is the tank itself. Random tank failures are negligibly small so tank failure is simply represented by a hazard-location failure, TKL.

Suction piping failure could be a random flow path failure (pipe or valves), SPR, or failure of the location containing it, Room A, or RMA.

Similarly, pump failure is PMPR or RMA.

AC power failure could be a random diesel generator failure DGR, a failure of the location containing it, RMB, or a failure of Room C through which the ac power cable runs from the diesel to the pump, RMC, (where random cable failure is neglected).

DC power has 2 sources, both of which must fail to fail dc power; (BAT or RMD) and (CHR + RMD + DGR + RMB + RME); that is, fail the battery or its location <u>and</u> the charger or its location or the diesel or its location (RMB) or the cable route location, Room E (RME).

Only random failure of the discharge flow path (DPR) need be considered since all its locations withstand speeds exceeding 250 mph (containment failure is expected at 250 mph).

Then system failure S, is:

S = Tank or suction pipe or ac or dc or pump or discharge pipe failure.

Then,

S = TKL + SPR + RMA + PMPR + RMA + DGR + RMB +
RMC + ((BAT + RMD) * (CHR + RMD + DGR + RMB + RME)) +
DPR

This logic expression could, of course, be reduced.

The following is a description, and, as required, logic expression development, for each top event of the three event trees to be quantified, Figures 3-4 through 3-6. The top events are ordered below alphabetically. All acronyms used in the logic expressions are defined in Appendix A.

Top Event A

This is the initiating event, loss of off-site power, which was discussed at length in Section 6.2.1.

Top Event B

Event B, "Atmospheric Steam Dump Availability", is considered in the analysis. In general, power is considered to be available to the atmospheric steam dump as long as Emergency 480 V Bus 1 or 3 is available and powered from their respective diesel generators and the NRV structure is intact. The cable continuity from the Emergency 480 V ac bus is assured for all areas from the NRVs back through the Main Control Room and Switchgear Rooms, since it is run through areas constructed of reinforced concrete. The only "soft" area is the North Wall of the Safety Injection Building where the cabling from the Emergency 480 V bus exits the building underground to the Turbine Building. Additionally, failure of the Cable Tray House was conservatively assumed to result in failure of Atmospheric Steam Dump availability even though no cabling related to these dump valves passes through the Cable Tray House; the reason being that Control Room instrumentation could be severely degraded by

Cable Tray House failure. Without instrumentation, it is unlikely that the operator would open the atmospheric steam dump valves within the five minutes required to prevent a PORV challenge.

Based on the above, a logic expression for Event B can be written:

B = NRVL + ((ASD12R + DC1R + DG1R + DG1L) *
(ASD34R + DC3R + SIB + DG3R + DG3L)) +
SIBN + SWGR + MCR + CBLT

Note that all acronyms used in logic expressions for this analysis are defined in Appendix A. Further detail is provided in Sections 6.4 and 6.5.

Top Event C

Event C, "Operator Actuation for Atmospheric Steam Dump Within 5 Minutes", is required to prevent a challenge of the PORV. This is an "immediate action" in the loss of off-site power procedure.

Top Event D

Event D, "Power to PORV Available", is dependent upon both random failure and the hazard intensity. The PORV is powered from the station battery bus in the Switchgear Room and actuated by the pressurizer pressure instrumentation which is also powered from the station battery bus. Once actuated, the dc power is routed to the PORV via the Main Control Room, Cable Tray House, and the Vapor Container. The "soft" area here is the Cable Tray House. The failure logic expression for Event D is:

D = DC1R + SWGR + CBLTPV + MCR

As discussed in Section 6.5, failure probability of the PORV power cable through the Cable Tray House (CBLTPV) is conservatively always taken as zero. Note that failure of Event D reduces the chance of a LOCA.

Top Event E

This event represents failure of the Power Operated Relief Valve (PORV) to open given a demand and available power. It is a simple component failure.

Top Event F

This event represents failure of the PORV to reclose once opened. A stuck open PORV is a LOCA, unless it can be isolated.

Top Event G

Event G, "Power to PORV Block Valve", considers that, given the PORV has actuated and fails to reclose, the operator attempts to close the PORV block valve. The event asks whether power is available to close the valve, given the operator attempts to close it. The operator has many sources of information available to him to indicate that the PORV has failed to close, and with the recent emphasis on PORV failures to close, it can be relatively well assured that the operator will attempt to close the block valve. The PORV block valve, PR-MOV-512, is supplied with emergency 480 V ac power from the Emergency Motor Control Center (EMCC) No. 1 in the Switchgear Room. Power is normally supplied to this EMCC from Emergency 480 V ac Bus No. 1. Power is then applied to the PORV block valve via cabling that runs from the Switchgear Room through the Main Control Room and Cable Tray House to the Vapor Container. The weak areas for this path are the SI Building North Wall and the Cable Tray House. Based on the above discussion, the logic expression representing failure of block valve power is:

G = DC1R + DG1R + DG1L + SIBN + SWGR + MCR + CBLT

Top Event H

"PORV Block Valve Closes" represents failure of the block valve to close given power and demand. It is simply a random valve failure. A LOCA results if the block valve fails to isolate a stuck open PORV.

Top Event I

At least one of the two primary code safety valves opens to relieve excessive Main Coolant System (MCS) pressure. This event is only challenged if neither the atmospheric steam dump nor the PORV has opened.

Top Event J

Primary code safety valve fails to reclose given that it has opened.

Top Event K

If the atmospheric steam dumps, the PORV and both safety valves fail to open, the loop safety valves will be challenged. Top Event K represents failure of all four loop safety valves to open. (Any one valve is sufficient to relieve excessive MCS pressure for this event).

Top Event L

Failure of one or more loop safety valves to reclose if opened would be a non-isolable LOCA.

Top Event LA

This is the LOCA initiating event. As previously discussed, the only LOCA to be reasonably considered for the wind/tornado hazard is a relief valve LOCA. Event LA is, then, entirely developed by the Relief Valve Challenges Event Tree, Figure 3-5.

Top Event LB

"Scram Occurs" represents the failure of the scram system to insert sufficient negative reactivity (via control rods). Scram failure is developed and discussed in great detail in the Yankee Nuclear Power Station Probabilistic Safety Study (YNPS PSS), Reference 3. The "Scram - No Cooldown" case from the PSS conservatively represents scram for this analysis.

Top Event LC

This event considers the control of positive reactivity insertion which could result from conditions such as excessive cooldown. It is discussed in detail in Section 6.2.2 as well as 3.2.1.

Top Event LD

This event considers safety injection into the MCS to mitigate the LOCA. Since, for this case, only "small" leakage is possible, any one train (of the three) of the Safety Injection System is considered sufficient as long as secondary heat removal is also successful (Events LF and LG). Note that safety injection could alternately be provided by the Charging System (all three trains); however, no credit has been taken. Failure of the SI System is represented by the logic expression:

SI LOCA = SIPIPE + SIT + SIB +
(SIP1R + DG1R + DG1L + DC1R + SWGR) *
(SIP2R + DG2R + DG2L + DC2R + SWGR) *
(SIP3R + DG3R + DG3L + DC3R)

Top Event LE

"Recirculation" is actually an extension of safety injection where suction valves are realigned to take water from the containment sump to form a recirculation cooling path. Then by quantifying SI through a long enough period to cover recirculation, the logic expression for the event can be written simply as:

RC = (DG1R + DG1L + RCMOV1) *
(DG3R + DG3L + RCMOV2) * OERCMOV

if credit is taken for remote or local operation of either of the two recirculation valves.

Top Event LF

"Feedwater Addition and Control" is discussed and developed under Event OE, below. LF is a reduced version of OE since credit is taken here for only the Safe Shutdown System and Steam-Driven Emergency Boiler Feedwater. Conservatively, SI, charging and all emergency ac power are assumed to be fully committed to Main Coolant System makeup.

The resulting, reduced logic expression for Event LF is:

FWLOCA = (ULPAB + SSSL + SSSR + FWST) *
(TK39 * TK1L + PPRM + ABR + SEBFP)

Top Event LG

This event examines "Steam Removal and Control" via the atmospheric steam dump valves or the secondary system's code safety valves. This is discussed further in Section 3.2.5.

Top Event LH

The availability of "Instrumentation Sufficient to Control" mitigative systems and monitor plant conditions during a LOCA is considered by this event. The instrumentation required during a LOCA is identical to that required for the loss of off-site power case (see Top Event OG) with the addition of safety injection pressure, flow, and tank level. Note that these additional parameters do not introduce any locations or important random failures not covered by the very conservative model for Event OG. For quantification purposes, it is reasonable to treat this Event (LH) and Top Event OG as the same event. (See also Sections 6.4 and 6.5).

Top Event OA

This top event examines the occurrence of a loss of off-site power event. If the hazard intensity is not severe enough to cause a loss of off-site power, no further analysis is performed. (See Section 6.2.1.)

Top Event OB

This top event examines the integrity of the Main Coolant System (MCS). If excessive leakage exists, a transfer to the LOCA event tree occurs. Section 6.2.3 explained the development of this top event.

Top Event OC

This top event examines reactivity control. Success is defined as insertion of greater than half of the control rods. It is an equivalent event to LB discussed previously in this section.

The reactivity control CSF consists of two major factors as discussed in Section 3.0: 1) insertion of negative reactivity and 2) control of positive reactivity addition. The positive reactivity control is provided by the MCS heat removal CSF; the negative reactivity control is provided by insertion of control rods or chemical shutdown.

To provide for control rod insertion (scram), the system must detect the need for scram and initiate a scram, and the rods must insert. Rod insertion is relatively well ensured since the rods fall in by gravity barring any binding that may cause a rod to stick. This is accounted for in the scram failure probability in the YNPS PSS. The detection of need to scram is provided by a diversity of sources both on the plant electrical system (generator relaying) and primary system (MCP undercurrent/overcurrent flow trip). These signals initiate tripping of relays in the control circuit of the rod control scram breakers, BK-1 and BK-2 from Battery Bus No. 1. The scram breakers' trip coil is then energized by Battery Bus No. 2 to initiate the scram. Even a loss of Battery Bus No. 1 initiates opening of the scram breakers due to loss of power to the MCP uc/oc cabinet. The detection circuits are located in the Main Control Room and vapor container; the battery buses and scram breakers are located in the Switchgear Room.

In the unlikely event that the rods fail to insert, chemical shutdown can be provided by one diesel generator feeding a 480 V non-IE bus to run a charging pump and opening the boric acid suction valve remotely from the Control Room (powered from MCC 4 via 480 V Bus 5-2) or manually in the Lower

Level Primary Auxiliary Building (LL PAB). This analysis takes no credit for chemical shutdown.

The wind/tornado effects on scram are negligible in that the Cable Tray House failure should induce a scram signal upon the loss of instrumentation. If a scram signal is not induced by the failure, the diversity of instrumentation will cause a scram signal to be affected. In the event that a scram signal is inhibited, the manual trip is still available to the operator at the Main Control Board. Actuation by the operator is assured since it is a routine, almost reflex action to manually initiate scram on <u>every</u> trip. If a battery bus should be failed by the event, scram is assured since loss of dc Bus No. 1 or 2 initiates a scram.

Top Event OD

This top event examines control of "positive reactivity insertion". It is equivalent to LC.

Top Event OE

This top event examines feedwater addition to the secondary. No credit is taken for direct "feed and bleed" cooling through the MCS.

Feedwater addition and control can be provided by any one of the following:

- o Electric-driven emergency boiler feedwater pump (either of 2) supplying through either the main feedwater header or the blowdown header.
- o The single steam-driven emergency boiler feedwater pump supplying through the main feedwater header.
- o All 3 charging pumps supplying through either the main feedwater header or the blowdown header.

- One safety injection pump train (of the 3) supplying through the blowdown header.

- The Safe Shutdown System supplying through the blowdown header.

For any of the above, a total feedwater flow of at least 100 gpm is required initially; this model conservatively assumes at least 2 steam generators must be supplied. The following water sources are available to the various systems:

	EEBF	SEBF	CHAR	SI	SSS
TK1	X	X			
TK39	X	X	X		
SIT			X	X	
FWST					X

Note that certain other suction and/or discharge flow paths are available in some cases; conservatively, no credit has been taken.

For all of the above, both random equipment failures (pumps, valves, pipe, power, etc.) and "location" failures caused by the hazard were considered.

The following set of logic expressions was used to represent failure to supply feedwater assuming no off-site power:

```
FW=CHAR*SI*EEBF*SEBF*SSS.
CHAR=3CHPP+3DG+3DC+(SIT*TK39)+(CHBLDN*CHMFWH).
3DG=DG1R+DG2R+DG3R+DG1L+DG2L+DG3L.
3DC=DC1R+DC2R+DC3R+SWGR+SIB.
SI=(SI1R+DG1R+DG1L+DC1R+SWGR)*(SI2R+DG2R+DG2L+DC2R+SWGR)*
   (SI3R+DG3R+DG3L+DC3R)+SIB+SIT+SIBLDN.
EEBF=(EBFP1R+DG1R+DG1L+DC1R+24V3R)*(EBFP2R+DG3R+DG3L+DC3R+24V2R+SIB)+
     SIBN+SWGR+LLPAB+(TK1*TK39)+(EBFBLDN*EBFMFWH).
SEBF=SEBFP+ABR+(TK1*TK39)+SEBFMFWH.
SSS=SSSBLDN+FWST+SSSR+SSSL.
CHBLDN=LLPAB+ULPAB.
CHMFWH=LLPAB+PPRM.
EBFBLDN=LLPAB+ULPAB.
```

SEBFMFWH=PPRM.

EBFMFWH=LLPAB+PPRM.

SIBLDN=ULPAB.

SSSBLDN=ULPAB.

TK1=TK1L+ABR.

Note that TK1 has two possible location failures, since a certain pipe break in the Auxiliary Boiler Room would result in draining of the tank.

Top Event OF

This top event examines steam removal through the secondary atmospheric steam dump valves or code safety valves. It is an equivalent event to LG discussed above.

Top Event OG

This top event examines the availability of instrumentation sufficient to monitor and control plant conditions. The following instrumentation was examined:

1. Core exit thermocouples

2. MCS pressure

3. Steam generator pressure

4. Steam generator level

5. Emergency feedwater flow

6. Safety and relief valve flow indicators

In order to properly control the parameters of the Main Coolant System heat removal, main coolant pressure control, and reactivity control critical safety functions, the operator must be able to assess these parameters for absolute values and trends. This is provided by instrumentation that is

either in the Main Control Room or remote from the Main Control Room which can be either installed or temporary depending upon the remaining cable continuity and power supplies following the event.

Instrumentation in the Main Control Room is that "normal" Main Control Room instrumentation that the operator has immediately available to read in the desired units. Instrumentation remote from the Main Control Room is that instrumentation which provides a redundant means of monitoring the critical parameters in an area such as local safety injection tank level or instrumentation available in, and powered from, the Safe Shutdown System facility. Installed instrumentation is that instrumentation that is already installed and can be read directly in the desired units by the operator. Temporary instrumentation is that instrumentation that is connected following an extreme event which may or may not read directly in the desired parameter, such as safety injection tank level read from a clear hose in inches that must be converted to gallons or core exit thermocouple readings taken by an Instrumentation Technician either in the SSS facility or directly at the containment penetration with a millivolt potentiometer that must be converted to degrees Fahrenheit.

The instrumentation considered critical to safe shutdown following an event resulting in a loss of off-site power is:

- o Pressurizer Wide-Range Level
- o Main Coolant System or Pressurizer Pressure
- o Main Coolant Temperature - Wide-Range
- o Core Exit Temperature
- o Steam Generator Level
- o Steam Generator Pressure

The instrumentation was modeled and reduced in the SETS computer code, quantified by the QUANTV code and input as a basic event to the loss of off-site power and LOCA event trees. Failure of any one parameter was set to failure of the system. For safety injection instrumentation, the additional instrument did not effect the failure probability significantly. The instrumentation failure expression is:

```
INS=(INSA*SSSI*LCLI)+SGPI.
INSA=INS1*INS2.
INS1=INST1+CBLT+PWRCH1.
PWRCH1=VB1R+SWGR+((EBS1R+DG1R+DG1L+DC1D+SIBN)
       *(UPS1R+DC1RUN)).
DC1RUN=DCBS1R+((BATCGR1R+MCC1BS1R+BS5-2R +
       EBS3R+DG3R+DG3L+DC3D)*(BAT1RUN+DC1L)).
INS2=INST2+CBLT+PWRCH2.
PWRCH2=VBS2R+SWGR+((EBS3R+DG3R+DG3L+DC3D+SIB+SIBN)*
       (UPS1R + DC2RUN))
DC2RUN=DCBS2R+((BATCGR2R+MCC2BS2R+BS6-3R+EBS1R+DG1R+DG1L+DC1D)*
       (BAT2RUN + DC2L)).
SSSI=SSSINST+SSSR+SSSL.
```

Note that Cable Tray House failure is modeled to fail all normal Main Control Room instrumentation. This approach is considered to be quite conservative since the cabling in the Cable Tray House is laid in "ladder-type" trays spaced one above the other at approximately 2-foot intervals. Since the cables are laid in the trays loosely and the block wall has little velocity when it strikes the cable tray, it is difficult to believe that even half of the cables in any tray would be broken or shorted. It would require an extremely elaborate model to predict what instrument cables have a probability of surviving and, therefore, failure of the concrete block cladding has been assumed to be failure of all cabling passing through the Cable Tray House, except that to the PORV; PORV power is always modeled as available since such a condition increases the likelihood of a LOCA.

6.4 Random Failure Data

This section discusses all random failure data used in this analysis to quantify event sequences. The discussion is broken into two parts as follows:

- Basic events contained in the logic expressions developed in Section 6.3 are discussed and assigned failure data in Section 6.4.1.

- o Top events not having logic expressions developed in Section 6.3 (either simple events or events already developed in the YNPS PSS) are assigned failure data in Section 6.4.2.

6.4.1 Fault Tree Basic Events

The logic expressions developed in Section 6.3 include certain basic events to represent failures of plant equipment and operator errors. These basic events, which are defined in Appendix A, and their failure probabilities, are listed in Table 6-2. The following paragraphs discuss each event and the basis of its failure probability as used in this analysis.

ASD12R(F): Atmospheric Steam Dump Valve 1 or 2 fails to open. From Table 6-1, MOV failure is 3.0×10^{-3} (open failure rate assumed same as closed) so for either of 2 valves (2) x $(3.0 \times 10^{-3}) = 6.0 \times 10^{-3}$.

ASD34R(F): Atmospheric Steam Dump Valve 3 or 4 fails to open - see ASD12R(F).

BATCGR1R: Random failure of Battery Charger Number 1. From YNPS PSS (Table 7-2), charger failure rate is 1.226×10^{-5} per hour for 24 hours = 2.94×10^{-4}.

BATCGR2R: Random failure of Battery Charger Number 2. See BATCGR1R.

BAT1RUN: Battery Number One failure to run for 24 hours. It is assumed that, without recharging, or load shedding by the operator, the batteries will not last for 24 hours. No credit is taken for the batteries to run (for any length of time) without charging. This event's failure probability is set to 1.0 which is very conservative.

BAT2RUN: Battery Number Two failure to run for 24 hours. See BAT1RUN.

BS5-2R: 480 V Bus 5-2 failure. Using the per hour bus failure rate from Table 7-2 of YNPS PSS, 0.37×10^{-6} hour x 24 hours = 8.88×10^{-6}.

BS6-3R: 480 V Bus 6-3 failure. See BS5-2R.

DCBS1R: DC Bus Number One, random failure. Again, using the per hour bus failure rate from YNPS PSS, Table 7-2, for 24 hours yields 8.88×10^{-6}.

DCBS2R: DC Bus Number Two, random failure. See DCBS1R.

DC1D: Battery Number One - demand failure. From YNPS PSS, Table 7-2, 3.61×10^{-4}. For conservatism use 3.61×10^{-3}.

DC1R: 125 V dc Bus Number One fails to supply power for 24 hours given that off-site power is unavailable. From YNPS PSS (Table 7-2) (Reference 3), battery failure is 3.6×10^{-4} per demand and (4.2×10^{-8} per hour x 24 hours) = 1×10^{-6}. Note that the logic models take no credit for dc power if emergency ac is lost. Also, no cross-tie is credited. Use 3×10^{-3} for dc bus failure.

DC2R: 125 V dc Bus Number 2. See DC1R.

DC3D: Battery Number 3, demand failure. See DC1D.

DC3R: 125 V dc Bus Number 3. See DC1R.

DG1R: Diesel Generator Number 1 fails to supply emergency ac power for 24 hours. Note that the logic models take no credit for a diesel if its dc starting power is not available. From Reference 3, Table 9-1, the probability of failure of one of three diesels to start or continue to run for 4 hours is 5.0×10^{-2}, so for one of one

$(5.01 \times 10^{-2})/3 = 1.67 \times 10^{-2}$. Then for 24 hours use 0.1 which is very conservative.

DG2R: Diesel Generator Number 2. See DG1R.

DG3R: Diesel Generator Number 3. See DG1R.

EBFP1R: Electric-driven Emergency Boiler Feedwater Pump Number 1 failure to start or continue to run for 24 hours or flow path related failure. Pump failure to start and run, from Table 7-2 of Reference 3, is $1.25 \times 10^{-3} + 24 (1.01 \times 10^{-5}) = 1.5 \times 10^{-3}$. From Table 9-1 of Reference 3, failure of electric and steam EBF is 4.8×10^{-6}. So, conservatively use 1×10^{-2}.

EBFP2R: Electric-driven Emergency Boiler Feedwater Pump Number 2. See EBFP1R.

EBS1R: 480 V Emergency Bus Number 1, random failure. As for the other buses discussed above, 8.88×10^{-6}. (See BS5-2R).

EBS3R: 480 V Emergency Bus Number 3, random failure. See EBS1R.

INST1: This is a "super component" representing the failure of any one of the following instruments which feed instrumentation channel number 1:

- o Main Coolant System Pressure (1 and 3), or,
- o Pressurizer Level, or,
- o Core Exit Temperatures, and Cold Leg Temperature (1 and 3), or
- o Steam Generator Level (1 or 3)

From Table 7-2 of the YNPS PSS, the failure rate per hour for general instrumentation is 2.66×10^{-6} or for 24 hours, 6.38×10^{-5}. Then the failure rate of the above super component for 24 hours is:

$$INST1 = (6.38 \times 10^{-5})^2 + (6.38 \times 10^{-5}) + (6.38 \times 10^{-5})^3 + 2\ (6.38 \times 10^{-5}) = 1.9 \times 10^{-4}$$

INST2: This "super component" is similar to INST1 except that, for channel 2 instrumentation, only one main coolant pressure channel is available and no core exit temperature channel is involved. The result is INST2 = $4(6.38 \times 10^{-5}) + (6.38 \times 10^{-5})^2 = 2.55 \times 10^{-4}$.

LCLI: Local Instrumentation. The probability of failure of the various local instrumentation channels throughout the plant which could be used in the event of remote instrumentation failure was set to 0.1. This value is not dominated by actual instrumentation failure but by the difficulty in using it due to location and, in some cases, indirect readings (i.e., reading millivolts on a thermocouple and converting to degrees F).

MCC1BS1R: Random failure of Motor Control Center Number 1, Bus Number 1. Again, random bus failure for a 24-hour period is 8.88×10^{-6}. (See BS5-2R).

MCC2BS2R: Random failure of Motor Control Center Number 2, Bus Number 2. See MCC1BS1R.

OERCMOV: Operator Error - fails to open at least one of two recirculation valves locally given they failed to operate remotely. Use 0.1 which is conservative, especially since for this event, the time period available is relatively long.

RCMOV1: Motor-operated recirculation valve 1 fails to open remotely given a signal. MOV failure is 3.0×10^{-3} per the Relief Valve Data Table. (Table 6-1)

RCMOV2: Motor-operated recirculation valve 2. See RCMOV1, above.

SEBFP: Steam-driven emergency boiler feed pump fails to deliver feedwater. Use 1×10^{-2} which, by engineering judgement and a knowledge of general failure data, is believed to be reasonable and conservative.

SGPI: Steam Generator Pressure Instrumentation. From YNPS PSS, Table 7-2, for pressure sensors, the failure rate is 1.043×10^{-5} per hour. So, the probability of failure of a pressure instrument over 24 hours is 2.5×10^{-4}. This analysis has assumed that feeding two steam generators is successful feedwater addition. Since both generators will behave in a very similar manner, indication of pressure in either of the two is sufficient. Failure of SGPI then requires failure of both of the two pressure indicators on the generators being fed. For conservatism, double the failure rate; then SGPI = $2(2.5 \times 10^{-4})^2 = 2.5 \times 10^{-7}$.

SIPIPE: Failure of safety injection piping or valves such that sufficient flow paths for SI are not available. From Reference 3, Table 9-1, failure probabilities of the SI System are on the order of 10^{-3}, so using 1.0×10^{-3} here for flow paths is conservative since SI pumps are added separately in the logic expressions.

SIP1R: Safety Injection Pump 1 random failure, from Reference 3, Table 7-2, SI pump failure to start and run for 24 hours is about 1.5×10^{-3}. Use 1×10^{-2} conservatively.

SIP2R: SI Pump 2 - See SIP1R.

SIP3R: SI Pump 3 - See SIP1R.

Note regarding SI: depending on specific plant conditions, injection (to primary or secondary) may be accomplished by HPSI alone, LPSI alone or may require LPSI boosting HPSI. One LPSI pump can boost at least 2 HPSI pumps. The model "safety injection pump" is specifically not designated high or low pressure but is intended to mean whichever pump or combination is required. The very conservative 10^{-2} failure probability covers this situation.

SI1R: Random failure of safety injection Train Number 1 to supply sufficient feedwater to the secondary. From Table 9-2 of Reference 3, this case has a failure probability of 6.14×10^{-2} which is very conservative for this analysis since it includes operator error and is based on the assumption that 2 of 3 SI trains are committed to primary injection leaving only 1 available for secondary feeding. Conservatively, use the 6.14×10^{-2}.

SI2R: SI Train 2 - See SI1R.

SI3R: SI Train 3 - See SI1R.

SSSI: Safe Shutdown System Instrumentation. The proposed Safe Shutdown System will have one channel of each of the following five indications as a minimum:

- o Main Coolant System Pressure
- o Pressurizer Level
- o Steam Generator Level
- o Main Coolant Temperature
- o Core Exit Temperature

Conservatively assuming that failure of any one indication is instrumentation failure, and using 6.38×10^{-5} for failure of an instrument for 24 hours (as discussed earlier), then $SSSI = 5\ (6.38 \times 10^{-5}) = 3.15 \times 10^{-4}$.

SSSR: Random failure of the Safe Shutdown System. Since the SSS is not yet fully designed, a complete failure probability determination cannot be made. 1×10^{-2} has been assumed and is believed to be somewhat conservative based on information available to date.

UPS1R: Random failure of uninterruptible power supply Number 1. From Table 7-2 of YNPS PSS, for a static inverter, the failure rate per hour is 1.22×10^{-5}, so for 24 hours = 2.94×10^{-4}.

UPS2R: Random failure of uninterruptible power supply Number 2. See UPS1R.

VB1R: Vital Bus Number One random failure. As discussed several times above, probability of bus failure over 24 hours is 8.88×10^{-6}.

VBS2R: Vital Bus Number Two random failure. See VB1R.

24V2R: Random failure of 2400 volt Bus Number 2. Since off-site power is not available, this bus must be powered by a diesel generator and back fed through the normal 480 V bus; failure probability would be about 5×10^{-3}. However, this bus is connected by bus bar to the station service transformer located outside in the switchyard. There is some chance that hazard-induced damage causes a short at the station service transformer which fails the 2400 V bus. Conservatively set bus failure probability to 0.5.

24V3R: 2400 volt bus 3. See 24V2R.

3CHPP: The Charging System supplies feedwater to the secondary (all three pumps required). From Table 9-2 of Reference 3, the failure probability is 8.23×10^{-2} which is conservative for this analysis since it includes operator error. Use 8.23×10^{-2}.

The failure probabilities developed here for basic events will be used in Section 6.6.2 to quantify event trees using the logic expressions developed in Section 6.3 as well as the top event failure data discussed below.

This approach was used in assessing the survivability of systems available to satisfy each of the critical safety functions modeled in the event trees. Failure to satisfy any one critical safety function results in core melt.

6.4.2 Top Event Failure Data

Each top event for which a logic expression was developed in Section 6.3, was provided failure data in Section 6.4.1. This section provides failure data for those top events which were not further developed as fault trees but are basic events.

Top Event Failure Data Development for the Relief Valve Challenge Event Tree

Top Event C, Operator Actuates Atmospheric Steam Dump

The probability of failure of this event was set to 0.1 to conservatively account for this proceduralized immediate operator action for any loss of off-site power.

Top Event E, Power-Operated Relief Valve Opens

As developed in Table 6-1, the probability of a PORV failing to open on demand is 1×10^{-3}.

Top Event F, Power-Operated Relief Valve Closes

As developed in Table 6-1, the probability of a PORV failing to close on demand is 2×10^{-2}.

Top Event H, PORV Block Valve Closes

As developed in Table 6-1, the probability of a Motor-Operated Valve (MOV) failing to close on demand is 3×10^{-3}.

Top Event I, Pressurizer Code Safety Valve (1 of 2) Opens

As developed in Table 6-1, the probability of a code safety valve failing to open on demand is 1×10^{-4}. The failure of both valves to open on demand was set to 1×10^{-5} conservatively.

Top Event J, Pressurizer Code Safety Valve Closes

As developed in Table 6-1, the probability of a code safety valve failing to close is 2×10^{-3}.

Top Event K, Main Coolant System Loop Safety Valve Opens

As developed in Table 6-1, the probability of a safety valve failing to open on demand is 1×10^{-4}.

Top Event L, Main Coolant System Loop Safety Valve Closes

As developed in Table 6-1, the probability of a safety valve failing to close on demand is 2×10^{-3}.

Top Event Failure Data Development for the LOCA Event Tree

Top Event LB, Scram Occurs

From the analyses performed in the YNPS PSS, the failure to scram and insert at least 12 of the 24 control rods is completely dominated by the failure of the two scram breakers (99.99%). The failure of detectors and logic circuits provided an insignificant contribution to the failure to scram because of redundancy of the RPS and diversity of signals received upon a loss of off-site power. The failure probability of the system including independent and common mode failures is 1×10^{-5}.

Additionally, as long as ac power is available, the operator can charge boric acid to the Main Coolant System to induce sufficient shutdown margin. The failure probability of one or more charging pumps to provide borated water to the primary for chemical shutdown from the BAMT is 2.44×10^{-2}. Therefore, for the ac power available case, the probability of failure to scram is failure of scram and failure of chemical injection ($1 \times 10^{-5} \times 2.44 \times 10^{-2} = 2.44 \times 10^{-7}$) neglecting operator error However, no credit was taken for chemical injection in this analysis. The probability of failure to scram is 1×10^{-5}.

Top Event LC, Positive Reactivity Control

As discussed in Sections 6.3, 6.2.2, and 3.2.1 this event is a negligible contributor to failure of this event tree.

Top Event LG, Steam Removal and Control

As discussed in Section 6.3 and 3.2.5, this event is a negligible contributor to failure of this event tree.

Top Event Failure Data Development for the Loss of Off-Site Power Event Tree

Top Event OC, Scram Occurs

See Top Event LB, above.

Top Event OD, Positive Reactivity Control

See Top Event LC, above.

Top Event OF, Steam Removal and Control

See Top Event LG, above.

6.5 Location Failure Data

From the systems identified for maintenance of critical safety functions, a list of critical locations was developed in Section 3.4. These areas were analyzed to determine ultimate wind and tornado loading capacities in Section 5.0.

From the models developed for the event trees and fault trees in Sections 3.0 and 6.0, the minimum critical areas were identified. Each of these areas was reviewed in detail to determine the "failure" wind/tornado speed. As can be seen from Table 5-1 and 5-2, any given area has a variety of failure windspeeds based upon which wall (or the roof) is considered to fail the critical equipment in that location.

Each location was reviewed by a team consisting of an Environmental Engineer, Systems Engineer, Structural Engineer, and Risk Assessment Engineer. The failure mode of each system for each location was identified and the structural component causing the system failure was identified. The location failure wind/tornado speed was then set to that speed at which the limiting structural component was predicted to fail. In general, this was the lowest windspeed of any structural component in a given location. Exceptions to this rule are identified in the following description. The exceptions generally occurred when there was no system within the area of the limiting component or a roof failure could not be found to impact a particular system. To determine if equipment could be impacted by a cladding failure, it was assumed that the clad (block) could only fall in the equivalent of the wall height from its footing. The acronym Table (Appendix A), list includes the minimum critical areas.

ABR - The steam driven emergency boiler feedwater pump and its suction, discharge, and steam supply piping are located in the Auxiliary Boiler Room. It was determined that the wind could not impact the interior wall and that structural failure of the exterior wall could not impact the pump. Failure of the exterior wall could fault the pump suction piping from TK-1 or TK-39; therefore, the limiting structure is the Auxiliary Boiler Room South Wall, T1J2. Then, using Table 5-2, the ABR fails at 122 mph wind or a 162 mph tornado.

CBLT and CBLTPV - The Cable Tray House failure can fail cables entering and exiting the vapor container. Structural cladding (block wall) failure can impact the cables and sever or short them. The reviewers recognized that the cable tray house cladding failure would not sever all the cables, which could result in a more severe event. In particular, the power cable to the power-operated relief valve may not fail but the power cable to the PORV block valve may fail. This damage results in the PORV opening with no ability to close the PORV block valve if the PORV should fail to close.

Because of this potential failure mode, Cable Tray House failure was broken into 2 parts: CBLT represents failure of the structure and all cables except power to the PORV; CBLTPV represents failure of the structure and the PORV cable. For any and all cases where the Cable Tray House wind or tornado failure speed is exceeded, CBLT is set to failure; CBLTPV is never failed. This is quite conservative as it assumes the worst case failure of the Cable Tray House.

Since the Cable Tray House wall failure limits the capacity of this location and the roof failure was not predicted to have any significant impact on the cabling within the Cable Tray House, wall failure wind/tornado speeds were used in this analysis. Therefore, CBLT is taken as failed for any hazard above 69 mph wind and 65 mph tornado (Table 5-2). Since this failure is expected to be a major contributor to instrumentation failure and core melt frequency, the Cable Tray House was also analyzed for a backfit design capacity to the 10^{-4} hazard frequency wind/tornado speed (110 mph). This backfit design would yield an ultimate capacity of 196 mph wind or 186 mph tornado.

DG1L, DG2L, DG3L - The Diesel Generator location failures were found to be dominated by wall failure impacting the diesel cooling water supply and cooling air supply. The Diesel Generator Building West Wall, that is the West Wall of Diesel Generator Cubicle No. 3, D11053, suffers structural cladding failure of the wall at a much lower windspeed than the building North Wall and therefore limits No. 3 cubicle. The possibility of interior West Wall failure of each successive diesel cubicle was reviewed by the team. It was determined that this failure was not very plausible since the time of passing of the tornado or high wind is so short as to not significantly impact the remaining (interior) West Walls. The failing of the building North Wall will cause the same failure mode, loss of cooling, but at a much higher windspeed and impacts all three diesel generators. The walls are D1V1, D1V2, and D1V3. Then, from Table 5-2, hazard-induced location failure of diesel generators 1 and 2 occur at 172 mph wind or 156 mph tornado; DG3L fails at 91 mph or 65 mph for wind or tornado, respectively.

FWST - The fire water storage tank is the sole supply credited for the Safe Shutdown System. Failure is taken as structural failure of the tank, which occurs at 190 mph (wind) or 182 mph (tornado).

LLPAB - Since the electric driven emergency boiler feed pumps, the electric supply (MCC) for Nos. 1 and 3 charging pumps, the suction piping to electric EBF and charging, as well as vapor container recirculation valves and piping are located in this common area, it was reviewed for failure mechanisms. It was determined that structural cladding failure of the North Wall, Wall Section P1E1 and P1E2, was the only feasible failure since the South Wall is made of reinforced concrete and is underground; the east and West Walls are bordered by adjacent rooms. Due to equipment locations within the area, failure of the North Wall could only impact the vapor container recirculation line and charging and electric EBF cross connect to main feedwater. All of these lines pass through the upper corner of the North Wall such that wall failure is unlikely to cause damage, especially to the larger (4") recirculation piping. For conservatism, however, LLPAB was included in the failure expressions for both electric emergency boiler feedwater and charging. LLPAB was not included in recirculation, however, as this would have been extremely overconservative. From Table 5-2, the limiting LLPAB wall, P1E1, fails at 197 or 222 mph for wind or tornado, respectively.

MCR - Main Control Room failure was included in the model for completeness though it is not predicted to fail for the spectrum of winds analyzed in this analysis since it is constructed of heavily reinforced concrete.

NRVL - The Non-Return Valve location was modeled since it may impact the excessive cooldown and steam removal events. In reviewing this area the team agreed that structural cladding failure may affect remote control of the atmospheric steam dump valves and non-return valves due to the possibility of severing electrical cables. This was not found to have any significant impact on excessive cooldown or steam removal since turbine throttle valve and control valve closure will protect the plant from excessive cooldown and the atmospheric steam dump valves can be operated by hand. In the unlikely event that the structural cladding failure inhibits access to the steam dump valves, the steam generator safety valves are assured of operation and result in successful steam removal. The limiting wind/tornado speed (Section 5.0) was determined from an extrapolated design windspeed for the structure, the uitimate capacity of this location is quite conservative since the limiting capacity of this structure is obviously static and dynamic loads from main steam piping. The non-return valve platform cladding is conservatively predicted to fail at 135 mph wind or 119 mph tornado.

SIBN - The Safety Injection Building North Wall was identified as a potentially critical area in that all cabling to and from the emergency diesel generators passes through the manhole and conduits on this wall. Though it is expected that the conduits, due to the vast number of them, would actually protect the wall from structural cladding failure. The cladding (block wall) failure is assumed to fault the cables at wall location D1X1. This wall is taken to fail at 134 mph and 121 mph for wind and tornado, respectively.

SIB - The Safety Injection Building has two walls, the South Wall and the West Wall, that could cause damage to Safety Injection System equipment, the No. 3 battery or the fire water tank heater (a potential flooding hazard). A review of the West Wall failure found that though the No. 3 train of safety injection pumps could be damaged, the remainder of the Safety Injection System would not be significantly impaired. Failure of the Safety Injection Building South Wall would fail all the LPSI pumps and the No. 3

battery resulting in failure of all three trains of SI and failure of Diesel No. 3 to start on demand. Therefore, failure of the Safety Injection Building was set to Safety Injection Building South Wall failure at wall sections D1Z1 and D1Z2. From Section 5.0, these walls are expected to fail at 103 mph wind or a 93 mph tornado.

SIT - The Safety Injection Tank provides a water source to the SI System and alternate source to the Charging System. Its failure was taken as structural failure of the tank, which is predicted to occur, for wind and tornados respectively, at 179 mph and 164 mph.

SSSL - The Safe Shutdown System Building was analyzed as the system will be installed to the Yankee composite seismic spectrum. The predicted ultimate windspeed capacities of the seismic design Safe Shutdown System are 258 mph for wind and 178 mph for tornados.

It is reiterated here that the original intent of this cost benefit analysis was to consider an upgraded SSS design to withstand wind/tornado hazards of 10^{-4} and 10^{-5} annual frequency. From Section 4.0 it is clear that 10^{-4} frequency is dominated by a 110 mph wind (95% confidence level) and 10^{-5} frequency is dominated by 165 mph tornado (95% confidence level).

Since the seismic design has an <u>ultimate</u> capacity of 258/178 mph (wind/tornado), an "upgraded" <u>design</u> capacity of 110 mph is not reasonable to evaluate. An upgrade to the 10^{-5} hazard, 165 mph design, would yield an ultimate capacity over 200 mph. This upgrade is worthy of a cost-benefit analysis. To maximize the potential benefits of such a backfit, the ultimate capacity was set to 250 mph for both wind and tornado.

SWGR - The Switchgear Room was recognized as a vital area for switchgear, buses, and dc power (Battery Rooms) and was included in the failure model to represent failure of any or all of these components. In the detailed review of fragilities, it was determined that:

1. The Battery Room walls T292 and T2G3 for Battery No. 2 or T2G4 for Battery No. 1 would dominate for wind/tornado capacity.

2. The exterior walls of the Switchgear Room would not enter into the model due to the inherent strength and location.

3. The Battery Room walls and roof are to be modified for seismic concerns and therefore, would no longer pose a problem once the backfit was complete, and

4. The Switchgear Room North Wall and the Battery Room walls are interior to the Turbine Building and even if Turbine Building cladding failure were to occur, they could not be directly impacted by high winds due to imposing turbine/condenser structure, feedwater heaters, and pipe whip/jet impingement plates.

The SWGR term was conservatively set to failure at the Battery Room No. 1 Wall failure wind/tornado speed for wall location T2G4 and the DC2R, random failure of dc Bus No. 2, was set to 1.0 at the wind/tornado speeds greater than the capacity of wall location T2G3.

From Table 5-2, it is clear that, for wind, SWGR failure is expected at 170 mph except for DC2 at 145 mph; for tornados, SWGR failure (including DC2) is predicted at 186 mph.

TK1L - The demineralized water tank failure location was taken as structural failure of the tank and entered in the model as such. Failure is predicted at 191 mph wind or 161 mph tornado.

TK-39 - The primary water storage tank failure was taken as structural failure of the tank. Section 5.0 predicts failure at 179 and 164 mph for wind and tornado, respectively.

ULPAB - The upper level Primary Auxiliary Building term was included in the model to account for location failure of the blowdown header either in the upper level PAB or non-radioactive (upper) pipe tunnel. The fragility analysis found that the ULPAB North Wall was dominant for the wind case and

the non-radioactive pipe tunnel was dominant for the tornado case. The model was quantified accordingly. Failure of the ULPAB roof was found to not be capable of impacting the blowdown header or connecting piping due to the location of the header (under the upper pipe tunnel) and imposing piping and tanks. The ultimate capacity of the non-radioactive pipe tunnel and ULPAB were determined assuming seismic backfits had already been completed. The capacities are 165 mph wind or 176 mph tornado.

For the remaining locations/critical areas identified in Section 3.4.2 each location was considered for location dependencies and the following reasoning was applied:

Station Service Transformer Yard - Since no specific data was available on the failure of the SST support structure the terms 24V2R and 24V3R were set to 0.5 to account for a 50/50 chance that given the yard structure is failed, it fails the 2400 V station service transformer bus to the Switchgear Room. (See Section 6.4.1).

Fuel Oil Tank - The fuel oil tank was not modeled since each diesel has its own 264 gallon fuel oil tank (day tank) and, assuming the diesel is running at full load, each diesel burns about 13 gallons per hour of fuel oil yielding a run time of about 20 hours. It is reasonable to assume that within 20 hours the operators could rig a temporary fuel supply or establish a bucket brigade to keep the diesel running, or provide an alternate feed path due to low decay heat levels.

Under VC - This area was not modeled since piping is not expected to be directly impacted by the windspeeds of concern in this analysis.

Pump Room - This location was not explicitly modeled since the piping of concern is located at the mezzenine level and is not expected to be impacted by any structural cladding (sheet metal) failure of the Turbine Building due to its own structural integrity (piping) and imposing beams, grits, and gratings.

Upper Level Primary Auxiliary Building West Wall - This area is not expected to impact the integrity of the safety injection and emergency feedwater piping even if it should fail due to the inherent structural integrity of the piping and other imposing piping and structures.

Primary Auxiliary Building Cubicle Area - This area was not modeled since it is constructed of reinforced concrete and will not be impacted by the winds of concern in this analysis.

South Yard - The south yard is covered by the two areas SSSL and FWST.

Turbine Building - The Turbine Building is not modeled since it was included to model the failure of the turbine and associated equipment. This was found to not be a concern since a structural failure in the area of the turbine can be anticipated to cause a turbine trip due to loss of control oil since the valves are spring driven to close.

Turbine Building West Staircase - This area was not modeled explicitly since failure of this area is not expected to directly impact any system credited in this analysis. The main purpose for including this area was to cover operator access for local manual operation of equipment. Since the human error probabilities modeled are conservatively high and there are many other ways to exit the Control Room, it was not considered necessary to explicitly model this area. This area is of concern for tornado venting because of ΔP concerns. This is a conservative concern since the walls of concern are inside walls of the Turbine Building. The cost estimate for the seismic backfit includes venting of this area.

Vapor Container - The Vapor Container is not included in the logic model since it is not anticipated to be impacted by any wind/tornado hazard below 250 mph. It is included as a separate input to the core melt and release probability for cost-benefit analysis. (Core melt and containment failure are assumed to occur for hazards exceeding 250 mph).

6.6 Core Melt Quantification

6.6.1 Mission Time

In order to assess the importance of random failures of plant equipment, it is necessary to establish a mission time for use in calculating equipment unreliability. The mission time selected for calculating system unreliabilities was 24 hours. The bases for the 24-hour mission time are discussed below.

Because this analysis is concerned with events involving a loss of off-site power, the key components include diesel generators. Other power supply equipment, such as electrical buses and battery charges, are minor contributors to electrical power reliability. As discussed in Section 6.4, diesel generator failure probability was set to a very conservative 0.1. In general, all random failure data used in this analysis is conservative. Furthermore, these component unreliabilities could be conservatively calculated based on long mission time requirements without significantly affecting the overall results.

The initiating event frequency (Section 6.2) is based on off-site power losses of durations less than 1 hour. Losses approaching or exceeding 24 hours are, then, much less likely than the event frequency used in this analysis. Also, as previously discussed, repair of off-site transmission equipment to restore power after significant damage could take on the order of a day.

The flow required to remove decay heat 24 hours after shutdown is about 30 gpm either to the core, if a LOCA has occurred, or to the steam generators for non-LOCA situations. This value is substantially lower than a typical plant because of the core size, equivalent to 600 Mwt at full power. If, at the end of one day after shutdown, all makeup systems failed, core uncovery would not occur for at least one-half day unless a LOCA existed. This time period allows for repair of equipment, re-energization of normal plant

equipment from off-site power, and use of portable makeup systems available at the plant.

Because makeup requirements are so low at 24 hours, less than 30 gpm, the success criteria are substantially less stringent than those modeled. This increases the number of plant systems available for success. For example, 2 or 3 charging pumps are required for a time period measured in hours after trip, whereas 1 charging pump is sufficient after 1 day.

For these reasons, a 24-hour mission time is reasonable for this analysis. In some cases, it is extremely conservative.

6.6.2 Event Tree Quantification

In order to support a cost-benefit analysis, the Event Trees must be quantified for four cases:

- The "base case", that is the present plant including modifications to be installed for the Yankee Composite Spectrum seismic upgrade. This includes the Safe Shutdown System.

- The "Cable Tray House Upgrade", which is the base case plus an upgraded Cable Tray House with a design windspeed of 110 mph (10^{-4} annual frequency).

- The "SSS Upgrade", which is the base case but with the new Safe Shutdown System designed to withstand a wind/tornado event of 10^{-5} annual frequency (165 mph).

- The "Combined Upgrade" which is the base case plus both the Cable Tray House upgrade and the SSS upgrade.

Recall that the original intent of this analysis was to consider SSS upgrades to 10^{-4} and 10^{-5} windspeeds. The SSS seismic design withstands a 10^{-4} wind without upgrade so there is no need to evaluate such a modification. The Cable Tray House case was added by the analysis team for

consideration since this area is expected to be an important contributor to the results of this core melt frequency calculation.

Additionally, the following conservatisms are important to understanding the core melt frequency quantification:

- o The present Cable Tray House is predicted to fail at 69 mph for wind and at 65 mph for tornados. (Diesel No. 3 location also fails at 65 mph tornado). The threshold speeds for loss of off-site power, the initiating event, are over 70 mph (Section 6.2.1). Model quantification will conservatively begin at cable tray failure speeds with off-site power assumed lost. (Note that this does not apply to the case of Cable Tray House Upgrade).

- o The tank failure speeds for wind used in the quantification are less than those actually predicted as follows:

	Failure Speed (MPH)	
Tank	Predicted	Used
SIT	179	163
TK1	191	174
TK39	179	163

Event tree quantification details are discussed below for each of the four cases.

Base Case

The event tree quantification was performed in two major parts, LOCA and non-LOCA, which were then summed along with common top events (scram and instrumentation). Both location and random failures must be considered.

Non-LOCA

This part considers the loss of off-site power event tree (Figure 3-4) where core melt occurrence is represented by the logic expression:

CM = OA * (OC + OD + OE + OF + OG)

Note that Event OB, failure of which leads to LOCA, not core melt, is developed by the relief valve event tree which is discussed below under LOCA.

Event OC (scram) and OG (instrumentation) are common with LOCA so they will be added separately. Top Events OD (positive reactivity control) and OF (steam removal) are negligible as discussed in Sections 3.2 and 6.4.2. Then this part of the quantification reduces to:

CM = OA * OE

Initiating event frequency for loss of off-site power was discussed in Sections 4.0 and 6.2. Event OE has a logic expression developed in Section 6.3 with appropriate data discussed in 6.4 and 6.5.

The OE logic expression was modeled and reduced using the SETS computer code and quantified with the code QUANTV. This was done in several stages to account for location failures at various windspeeds. Specifically, the hazard wind/tornado speeds were divided into several ranges with appropriate locations failed in each range. (Based on Section 6.5 data). Then, using the "OMEGA" option in SETS, reduced logic expressions were developed for each hazard range with appropriate locations forced to failure. Each expression was then quantified with QUANTV to account for random failures. The result is a conditional probability of core melt given the occurrence of a hazard within each range. From Section 4.0, the frequency of a hazard within that range can be determined. For each hazard speed range, multiplying the conditional core melt probability by the hazard frequency gives the core melt frequency for that range. Summing these products yields the total annual frequency of core melt due to feedwater failure for wind/tornado hazards. The following tables provide the details.

For the wind hazard (95% confidence level):

Speed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
69-91	$(3 \times 10^{-2}) - (1.8 \times 10^{-3})$	2.8×10^{-2}	2.91×10^{-7}	8.1×10^{-9}
91-103	$(1.8 \times 10^{-3}) - (3 \times 10^{-4})$	1.5×10^{-3}	2.21×10^{-6}	3.3×10^{-9}
103-122	$(3 \times 10^{-4}) - (2.5 \times 10^{-5})$	2.8×10^{-4}	6.13×10^{-5}	1.7×10^{-8}
122-135	$(2.5 \times 10^{-5}) - (2 \times 10^{-6})$	2.3×10^{-5}	6.12×10^{-3}	1.4×10^{-7}
135-145	$(2 \times 10^{-6}) - (2.5 \times 10^{-7})$	1.8×10^{-6}	1.00×10^{-2}	1.8×10^{-8}
145-163	$(2.5 \times 10^{-7}) - (10^{-8})$	1.5×10^{-7}	1.00×10^{-2}	$<1.5 \times 10^{-9}$
163-	$(<10^{-8})$ -	$<10^{-8}$	1.00	$<10^{-8}$

So, for the wind hazard, at the 95% confidence level, the total core melt frequency due to feedwater failure (non-LOCA) is about 1.9×10^{-7}.

For the wind hazard (50% confidence level):

Speed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
69-91	$(1 \times 10^{-2}) - (2 \times 10^{-4})$	9.8×10^{-3}	2.91×10^{-7}	2.9×10^{-9}
91-103	$(2 \times 10^{-4}) - (2.5 \times 10^{-5})$	1.8×10^{-4}	2.21×10^{-6}	4.0×10^{-10}
103-122	$(2.5 \times 10^{-5}) - (4 \times 10^{-7})$	2.5×10^{-5}	6.13×10^{-5}	1.5×10^{-9}
122-135	$(4 \times 10^{-7}) - (1 \times 10^{-8})$	3.9×10^{-7}	6.12×10^{-3}	2.4×10^{-9}
135-145	$(1 \times 10^{-8}) - (<10^{-8})$	ϵ	1.00×10^{-2}	$<10^{-10}$
145-163	$(<10^{-8}) - (<10^{-8})$	ϵ	1.00×10^{-2}	$<10^{-10}$
163-	$(<10^{-8})$ -	$<10^{-8}$	1.00	$<1 \times 10^{-8}$

Then, for the 50% confidence level wind hazard, the total core melt frequency due to feedwater failure (non-LOCA) is less than 10^{-8}.

For the tornado hazard (95% confidence level):

peed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
65-93	$(1.7 \times 10^{-4}) - (8.5 \times 10^{-5})$	8.5×10^{-5}	2.21×10^{-6}	1.9×10^{-10}
93-120	$(8.5 \times 10^{-5}) - (4 \times 10^{-5})$	4.5×10^{-5}	6.13×10^{-5}	2.8×10^{-9}
120-156	$(4 \times 10^{-5}) - (1.6 \times 10^{-5})$	2.4×10^{-5}	1.00×10^{-4}	2.4×10^{-9}
156-162	$(1.6 \times 10^{-5}) - (1.1 \times 10^{-5})$	5.0×10^{-6}	1.00×10^{-4}	5.0×10^{-10}
162-176	$(1.1 \times 10^{-5}) - (7.0 \times 10^{-6})$	4.0×10^{-6}	1.00×10^{-2}	4.0×10^{-8}
176-	(7.0×10^{-6}) -	7.0×10^{-6}	1.00	7.0×10^{-6}

he total core melt frequency due to (non-LOCA) feedwater failure is 7.0×10^{-6} for the 95% onfidence tornado hazard.

And for the tornado hazard (50% confidence level):

Speed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
65-93	$(4.5 \times 10^{-5}) - (2 \times 10^{-5})$	2.5×10^{-5}	2.21×10^{-6}	5.5×10^{-11}
93-120	$(2 \times 10^{-5}) - (7 \times 10^{-6})$	1.3×10^{-5}	6.13×10^{-5}	8.0×10^{-10}
120-156	$(7 \times 10^{-6}) - (2.0 \times 10^{-6})$	5.0×10^{-6}	1.00×10^{-4}	5.0×10^{-10}
156-162	$(2.0 \times 10^{-6}) - (1.7 \times 10^{-6})$	3.0×10^{-7}	1.00×10^{-4}	3.7×10^{-11}
162-176	$(1.7 \times 10^{-6}) - (9.2 \times 10^{-7})$	7.8×10^{-7}	1.00×10^{-2}	7.8×10^{-9}
176-	(9.2×10^{-7}) -	9.2×10^{-7}	1.00	9.2×10^{-7}

Then, for the 50% confidence level tornado hazard, the total core melt frequency due to (non-LOCA) feedwater failure is 9.3×10^{-7}.

To summarize the non-LOCA feedwater failure part:

	Core Melt Frequency	
Hazard	50% Hazard Confidence	95% Hazard Confidence
Wind	$<10^{-8}$	1.9×10^{-7}
Tornado	9.3×10^{-7}	7.0×10^{-6}

LOCA

The quantification method here is the same as for the non-LOCA case but the logic model is more involved. As previously discussed, the only LOCA which is reasonable to postulate for this analysis is one caused by relief valve failure. The relief valve event tree (Figure 3-5), an expansion of Event OB, is then the initiating Event LA for the LOCA Event Tree (Figure 3-6). A boolean logic expression for LOCA leading to core melt is:

$$CM = LA * (LB + LC + LD + LE + LF + LG + LH)$$

Again, scram (LB) and instrumentation (LH) will be added separately; positive reactivity control (LC) and steam removal (LG) are negligible. Logic expressions for safety injection (LD) and recirculation (LE) were developed in Section 6.3 as was a reduced feedwater for LOCA expression (LF). LA can be replaced by an expression for the relief valve tree leading to occurrence of a leak. This expression, in reduced form, is:

$$LA = (C + B) * (F * (H + G) + (E + D) * (J + I * L))$$

The top events within this expression were discussed in Sections 6.3 and 6.4.

The SETS computer code was used to merge this set of logic expressions, then substitute and reduce to one expression which represents the occurrence of <u>and</u> failure to mitigate a LOCA. (Excluding, of course, scram and instrumentation). The merged logic model can then be quantified in the same manner as the non-LOCA logic expression.

For wind at 95% hazard confidence level:

eed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
69-91	$(3 \times 10^{-2}) - (1.8 \times 10^{-3})$	2.8×10^{-2}	7.24×10^{-5}	2.0×10^{-6}
91-103	$(1.8 \times 10^{-3}) - (3 \times 10^{-4})$	1.5×10^{-3}	4.84×10^{-4}	7.3×10^{-7}
103-	(3×10^{-4}) -	3.0×10^{-4}	2.0×10^{-2}	6.0×10^{-6}

ne total annual core melt frequency due to relief valve LOCA due to a wind event (95% hazard onfidence) is 8.7×10^{-6}.

At 50% confidence for wind:

peed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
69-91	$(1 \times 10^{-2}) - (2 \times 10^{-4})$	9.8×10^{-3}	7.24×10^{-5}	7.1×10^{-7}
91-103	$(2 \times 10^{-4}) - (2.5 \times 10^{-5})$	1.8×10^{-4}	4.84×10^{-4}	8.7×10^{-8}
103-	(2.5×10^{-5}) -	2.5×10^{-5}	2.00×10^{-2}	5.0×10^{-7}

Then, for the wind hazard, at the 50% confidence level, the total core melt frequency due to relief valve LOCA is 1.3×10^{-6}.

For tornados at 95% hazard confidence,

Speed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
65-93	$(1.7 \times 10^{-4}) - (8.5 \times 10^{-5})$	8.5×10^{-5}	4.8×10^{-4}	4.1×10^{-8}
93-	(8.5×10^{-5}) -	8.5×10^{-5}	2.00×10^{-2}	1.7×10^{-6}

For the tornado hazard, at the 95% confidence level, the total core melt frequency due to relief valve LOCA is 1.7×10^{-6}.

And finally for tornados with a 50% hazard frequency confidence,

Speed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
65-93	$(4.5 \times 10^{-5}) - (2 \times 10^{-5})$	2.5×10^{-5}	4.84×10^{-4}	1.2×10^{-8}
93-	(2×10^{-5}) -	2.0×10^{-5}	2.00×10^{-2}	4.0×10^{-7}

For the 50% hazard confidence tornado, the total core melt frequency due to relief valve LOCA is 4.1×10^{-7}.

To summarize the LOCA failure (base case):

	Core Melt Frequency	
Hazard	50% Hazard Confidence	95% Hazard Confidence
Wind	1.3×10^{-6}	8.7×10^{-6}
Tornado	4.1×10^{-7}	1.7×10^{-6}

CABLE TRAY HOUSE UPGRADE

The ultimate failure speeds of a Cable Tray House designed to 110 mph (10^{-4} annual frequency) are predicted to be 196 mph wind and 186 mph tornado.

From the hazard curve,

Speed	Exceedance Frequency			
	50%		95%	
MPH	Wind	Tornado	Wind	Tornado
186	--	6.5×10^{-7}	--	5.0×10^{-6}
196	$<10^{-8}$	--	$<10^{-8}$	--

A logic model review indicates clearly that the Cable Tray House does not impact non-LOCA feedwater but does affect relief valve LOCA as well as instrumentation.

Instrumentation, and the relief valve LOCA are considered below for the upgraded Cable Tray House.

For Instrumentation:

Conservatively, no credit is taken for batteries if there is no ac power available to charge them. Then instrumentation failure probability is 10^{-5} with no hazard failures; 10^{-3} when all ac power or the Cable Tray House are lost and 10^{-1} when the Safe Shutdown System is also lost. (See Sections 6.3 through 6.5).

For wind events, ac power is lost at 135 mph when the SI Building North Wall (SIBN) fails; the SSS can withstand at least 250 mph wind.

For tornados, ac power fails with the SIBN at 120 mph; the SSS fails at 78 mph.

So for wind at 95% confidence level:

Speed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
69-91	$(3 \times 10^{-2}) - (1.8 \times 10^{-3})$	2.8×10^{-2}	10^{-5}	2.8×10^{-7}
91-135	$(1.8 \times 10^{-3}) - (2 \times 10^{-6})$	1.8×10^{-3}	10^{-4}	1.8×10^{-7}
135-	(2×10^{-6}) -	2.0×10^{-6}	10^{-3}	2.0×10^{-9}

For tornados at the 95% level:

Speed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
65-120	$(1.7 \times 10^{-4}) - (4.0 \times 10^{-5})$	1.3×10^{-4}	10^{-4}	1.3×10^{-8}
120-178	$(4.0 \times 10^{-5}) - (7.0 \times 10^{-6})$	3.3×10^{-5}	10^{-3}	3.3×10^{-8}
178-	(7.0×10^{-6}) -	7.0×10^{-6}	10^{-1}	7.0×10^{-7}

Then, for the combined wind/tornado hazard at the 95% confidence level the total core melt frequency due to instrumentation failure, is 1.2×10^{-6}, if the Cable Tray House is upgraded to 110 mph design.

Now, for wind at the 50% level:

eed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
69-91	$(1 \times 10^{-2}) - (2 \times 10^{-4})$	1.0×10^{-2}	10^{-5}	1.0×10^{-7}
91-135	$(2 \times 10^{-4}) - (1 \times 10^{-8})$	2.0×10^{-4}	10^{-4}	2.0×10^{-8}
135-	(1×10^{-8}) -	10^{-8}	10^{-3}	1.0×10^{-11}

And for tornados at the 50% confidence level:

peed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
65-120	$(4.5 \times 10^{-3}) - (7.0 \times 10^{-6})$	3.8×10^{-5}	10^{-4}	3.8×10^{-9}
120-178	$(7.0 \times 10^{-6}) - (9.2 \times 10^{-7})$	6.1×10^{-6}	10^{-3}	6.1×10^{-9}
178-	(9.2×10^{-7})	9.2×10^{-7}	10^{-1}	9.2×10^{-8}

hen for the combined wind/tornado hazard at the 50% confidence level, the total core melt requency due to instrumentation failure, is 2.2×10^{-7}, if the Cable Tray House is pgraded to 110 mph design.

<u>For relief valve LOCA</u> with the Cable Tray House 110 mph design modification:

At the 95% hazard confidence for wind:

Speed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
77-91	$(1 \times 10^{-2}) - (1.8 \times 10^{-3})$	8.2×10^{-3}	4.49×10^{-5}	3.7×10^{-7}
91-103	$(1.8 \times 10^{-3}) - (3 \times 10^{-4})$	1.5×10^{-3}	4.36×10^{-4}	6.5×10^{-7}
103-122	$(3 \times 10^{-4}) - (2.5 \times 10^{-5})$	2.8×10^{-4}	2.07×10^{-3}	5.8×10^{-7}
122-135	$(2.5 \times 10^{-5}) - (2.0 \times 10^{-6})$	2.3×10^{-5}	2.07×10^{-3}	4.8×10^{-8}
135-	(2.0×10^{-6}) -	2.0×10^{-6}	2.00×10^{-2}	4.0×10^{-8}

For the wind hazard, at the 95% confidence level, the total core melt frequency due to relief valve LOCA excluding instrumentation failure is 1.7×10^{-6}, if the Cable Tray House is upgraded to the 110 mph design.

At the 95% confidence level for tornados:

peed Interval (MPH)	Point Frequency (95% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
70-93	$(2.0 \times 10^{-4}) - (8.5 \times 10^{-5})$	1.2×10^{-4}	4.36×10^{-4}	5.2×10^{-8}
93-120	$(8.5 \times 10^{-5}) - (4.0 \times 10^{-5})$	4.5×10^{-5}	2.07×10^{-3}	9.3×10^{-8}
120-	(4.0×10^{-5}) -	4.0×10^{-5}	2.00×10^{-2}	8.0×10^{-7}

or the tornado hazard, at the 95% confidence level, the total core melt frequency due to elief valve LOCA, excluding instrumentation failure is 9.5×10^{-7}, if the Cable Tray House s modified to 110 mph design.

At the 50% hazard level for wind:

Speed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
75-91	$(4 \times 10^{-3}) - (2.0 \times 10^{-4})$	3.8×10^{-3}	4.49×10^{-5}	1.7×10^{-7}
91-103	$(2.0 \times 10^{-4}) - (2.5 \times 10^{-5})$	1.8×10^{-4}	4.36×10^{-4}	7.8×10^{-8}
103-122	$(2.5 \times 10^{-5}) - (4.0 \times 10^{-7})$	2.5×10^{-5}	2.07×10^{-3}	5.2×10^{-8}
122-135	$(4 \times 10^{-7}) - (1 \times 10^{-8})$	3.9×10^{-7}	2.07×10^{-3}	8.1×10^{-10}
135-	(1×10^{-8}) -	1×10^{-8}	2.00×10^{-2}	2.0×10^{-10}

For the wind hazard, at the 50% confidence level, the total core melt frequency due to relief valve LOCA, excluding instrumentation failure is 3.0×10^{-7} if the Cable Tray House is upgraded to the 110 mph design.

At the 50% level for tornado:

peed Interval (MPH)	Point Frequency (50% Confidence)	Interval Frequency	Core Melt Probability	Interval Core Melt Frequency
70-93	$(4 \times 10^{-5}) - (2 \times 10^{-5})$	2.0×10^{-5}	4.36×10^{-4}	8.7×10^{-9}
93-120	$(2 \times 10^{-5}) - (7 \times 10^{-6})$	1.3×10^{-5}	2.07×10^{-3}	2.7×10^{-8}
120-	(7×10^{-6}) -	7.0×10^{-6}	2.00×10^{-2}	1.4×10^{-7}

or the tornado hazard, at the 50% confidence level, the total core melt frequency due to elief valve LOCA, excluding instrumentation failure is 1.8×10^{-7}, if the Cable Tray house s modified to 110 mph design.

SSS UPGRADE

If the Safe Shutdown System design is upgraded to 165 mph design windspeed (10^{-5} event), its ultimate failure speeds will exceed 200 mph. In order to maximize benefit of this upgrade (conservative for cost benefit), it is assumed here that the upgraded system will withstand 250 mph. Note that the upgrade must include the Fire Water Storage Tank and the Upper Level Primary Auxiliary Building to be effective.

This upgrade has no effect on LOCA since for the "base case" plant, the SSS survives a stronger hazard than safety injection. Feedwater and instrumentation are both affected by an improved SSS capacity.

Feedwater

There is no change below 163 mph wind or 176 mph tornado, since for the "base case" plant, no damage to any SSS related location occurs below these speeds. Then, for wind 163 to 250 mph melt probability becomes 10^{-2} improving from 1.0; interval core melt frequency (95%) goes from less than 10^{-8} to less than 10^{-10}. For this case, total core melt frequency remains unchanged at 1.9×10^{-7}. Applying the same at the 50% level, the total core melt frequency again remains unchanged at less than 10^{-8}.

Now consider tornados for feedwater on the 95% confidence level. Above 176 mph the core melt probability changes from 1.0 to 10^{-2} so the interval core melt frequency improves to 7.0×10^{-8} from 7.0×10^{-6}. Total core melt frequency due to feedwater failure becomes 1.2×10^{-7} for the 95% confidence tornado hazard.

At the 50% level, interval melt frequency above 176 mph, becomes 9.2×10^{-9} resulting in a total frequency of 1.8×10^{-8} which is an improvement from 9.3×10^{-7}.

Instrumentation

Instrumentation failure probability is not significantly improved by this upgrade such that the value remains unchanged from the base case at 10^{-3}.

COMBINED UPGRADE

Finally, look at both upgrades together (110 mph Cable Tray House design and 165 mph Safe Shutdown System design).

Feedwater System results for the SSS upgrade are appropriate for this case; LOCA results for the Cable Tray House upgrade are also appropriate. Instrumentation for this combined upgrade is an improvement over the Cable tray case since here, for tornados, instrumentation never gets to 10^{-1}. (SSS does not fail below 250 mph). Then, for tornados at the 95% level, interval core melt frequency above 178 mph goes from 7.0×10^{-7} to 7.0×10^{-9} so for the combined wind/tornado hazard, total frequency due to instrumentation improves from 1.2×10^{-6} to 5.2×10^{-7}.

At the 50% confidence level, interval core melt frequency above 178 mph goes from 9.2×10^{-8} to 9.2×10^{-10} improving the combined wind/tornado total frequency of core melt due to instrumentation failure from 2.2×10^{-7} to 1.3×10^{-7}.

Then, to summarize event tree quantification results for each of the four cases,

For 95% hazard confidence:

Case	Core Melt Frequency Per Year Due To: Instrumentation Wind+Tornado	Non-LOCA Wind/Tornado	Relief Valve LOCA Wind/Tornado
Base	3.0×10^{-5}	$1.9 \times 10^{-7}/7.0 \times 10^{-6}$	$8.7 \times 10^{-6}/1.7 \times 10^{-6}$
Cable Tray House Upgrade	1.2×10^{-6}	$1.9 \times 10^{-7}/7.0 \times 10^{-6}$	$1.7 \times 10^{-6}/9.5 \times 10^{-7}$
SSS Upgrade	3.0×10^{-5}	$1.9 \times 10^{-7}/1.2 \times 10^{-7}$	$8.7 \times 10^{-6}/1.7 \times 10^{-6}$
Combined Upgrade	5.2×10^{-7}	$1.9 \times 10^{-7}/1.2 \times 10^{-7}$	$1.7 \times 10^{-6}/9.5 \times 10^{-7}$

For the 50% Hazard Confidence:

Case	Core Melt Frequency Per Year Due to: Instrumentation Wind+Tornado	Non-LOCA Wind/Tornado	Relief Valve LOCA Wind/Tornado
Base	1.0×10^{-5}	$<10^{-8}/9.3 \times 10^{-7}$	$1.3 \times 10^{-6}/4.1 \times 10^{-7}$
Cable Tray House Upgrade	2.2×10^{-7}	$<10^{-8}/9.3 \times 10^{-7}$	$3.0 \times 10^{-7}/1.8 \times 10^{-7}$
SSS Upgrade	1.0×10^{-5}	$<10^{-8}/1.8 \times 10^{-8}$	$1.3 \times 10^{-6}/4.1 \times 10^{-7}$
Combined Upgrade	1.3×10^{-7}	$<10^{-8}/1.8 \times 10^{-8}$	$3.0 \times 10^{-7}/1.8 \times 10^{-7}$

The next section combines the above information to determine total core melt frequency.

6.6.3 Overall Results

The total annual core melt frequency due to the wind/tornado hazard is determined by simply adding the Event Tree results to the scram and instrumentation failure frequency.

Scram, from Section 6.4.2 has a failure probability of 1.0×10^{-5} which is constant over the hazard speed range and LOCA/non-LOCA events. The dominate event threshold frequencies are for wind, being 3×10^{-2} and 1×10^{-2} for 50% and 95% confidence levels, respectively. Then, core melt frequency due to scram failure is 3×10^{-7} (95%) or 1×10^{-7} (50%). Note that it is conservatively assumed that scram failure leads to core melt.

Instrumentation is as discussed in Sections 6.3 (base case) and 6.6.2 (upgrade cases).

Overall results are summed as follows:

For the base case and 95% hazard confidence:

Non-LOCA feedwater	-	wind	1.9×10^{-7}
	-	tornado	7.0×10^{-6}
Relief Valve LOCA	-	wind	8.7×10^{-6}
	-	tornado	1.7×10^{-6}
Scram			3.0×10^{-7}
Instrumentation			3.0×10^{-5}
Total annual core melt frequency			4.8×10^{-5}

where instrumentation is clearly the major contributor here being 62.5% of the total. Recall that Cable Tray house failure is the major contributor to instrumentation failure probability.

For the base case and 50% hazard confidence:

Non-LOCA feedwater	-	wind	$<10^{-8}$
	-	tornado	9.3×10^{-7}
Relief Valve LOCA	-	wind	1.3×10^{-6}
	-	tornado	4.1×10^{-7}
Scram			1.0×10^{-7}
Instrumentation			1.0×10^{-5}
Total annual core melt frequency			1.3×10^{-5}

where instrumentation contributes 77% at the 50% hazard confidence.

If the Cable Tray House is upgraded to 110 mph design, the overall results improve as follows.

For 95% hazard confidence:

Non-LOCA feedwater	-	wind	1.9×10^{-7}
	-	tornado	7.0×10^{-6}
Relief Valve LOCA	-	wind	1.7×10^{-6}
	-	tornado	9.5×10^{-7}
Scram			3.0×10^{-7}
Instrumentation			1.2×10^{-6}
Total annual core melt frequency			1.1×10^{-5}

Instrumentation contributes only about 11% here with the main contributor being non-LOCA feedwater for the tornado hazard (64%).

Considering the Cable Tray House upgrade at 50% hazard confidence level:

Non-LOCA feedwater	-	wind	$<10^{-8}$
	-	tornado	9.3×10^{-7}
Relief Valve LOCA	-	wind	3.0×10^{-7}
	-	tornado	1.8×10^{-7}
Scram			1.0×10^{-7}
Instrumentation			2.2×10^{-7}
Total annual core melt frequency			1.7×10^{-6}

Instrumentation is about 13% of this total with non-LOCA feedwater for the tornado hazard being almost 55%.

For the case of the Safe Shutdown System upgrade (95% hazard confidence):

Non-LOCA feedwater	-	wind	1.9×10^{-7}
	-	tornado	1.2×10^{-7}
Relief Valve LOCA	-	wind	8.7×10^{-6}
	-	tornado	1.7×10^{-6}
Scram			3.0×10^{-7}
Instrumentation			3.0×10^{-5}
Total annual core melt frequency			4.1×10^{-5}

For the 50% hazard confidence with the Safe Shutdown System designed to 165 mph hazard:

Non-LOCA feedwater	-	wind	$<10^{-8}$
	-	tornado	1.8×10^{-8}
Relief Valve LOCA	-	wind	1.3×10^{-6}
	-	tornado	4.1×10^{-7}
Scram			1.0×10^{-7}
Instrumentation			1.0×10^{-5}
Total annual core melt frequency			1.2×10^{-5}

Finally, consider the combined modification case:

For the 95% hazard confidence case,

Non-LOCA feedwater	- wind	1.9×10^{-7}
	- tornado	1.2×10^{-7}
Relief Valve LOCA	- wind	1.7×10^{-6}
	- tornado	9.5×10^{-7}
Scram		3.0×10^{-7}
Instrumentation		5.2×10^{-7}
Total annual core melt frequency		3.8×10^{-6}

And for the 50% confidence combined upgrade:

Non-LOCA feedwater	- wind	$<10^{-8}$
	- tornado	1.8×10^{-8}
Relief Valve LOCA	- wind	3.0×10^{-7}
	- tornado	1.8×10^{-7}
Scram		1.0×10^{-7}
Instrumentation		1.3×10^{-7}
Total annual core melt frequency		7.3×10^{-7}

To summarize the overall core melt frequency results:

Total Core Melt Frequency Due to Wind/Tornado

Case	95% Confidence	50% Confidence
Base Case	4.8×10^{-5}	1.3×10^{-5}
Cable Tray House Upgrade	1.1×10^{-5}	1.7×10^{-6}
SSS Upgrade	4.1×10^{-5}	1.2×10^{-5}
Combined Upgrade	3.8×10^{-6}	7.3×10^{-7}

To put the potential modifications in some perspective, the following table presents each, in terms of reduction in core melt frequency, (at the 95% confidence level).

Improvement		Melt Frequency Reduced		Annual Core Melt Frequency Reduction
From Plant Condition	Proposed Upgrade	From	To	
Base Case	Cable Tray	4.8×10^{-5}	1.1×10^{-5}	3.7×10^{-5}
Base Case	SSS	4.8×10^{-5}	4.1×10^{-5}	0.7×10^{-5}
Base Case	Combined	4.8×10^{-5}	3.8×10^{-6}	4.4×10^{-5}
Cable Tray	SSS*	1.1×10^{-5}	3.8×10^{-6}	0.7×10^{-5}

*This case considers installation of the 10^{-5} SSS upgrade given that the 110 mph design Cable Tray House has been installed. The result, of course, is the combined upgrade.

The combined upgrade, of course, yields the maximum reduction in annual core melt frequency. It is important to note that 84% of this reduction can be accomplished by the Cable Tray House upgrade alone.

Core melt frequency alone is not necessarily a good indicator of plant risk and, therefore, cannot reliably indicate modification benefit. Further, any potential benefit must be weighed against its costs if an upgrade justification is to be valid. Section 9.0 provides further case comparison from a cost-benefit perspective.

6.7 Release Frequency

Section 6.6 determined annual core melt frequency for hazards up to 250 mph. The annual, release frequency for hazards up to 250 mph is the product of:

- o Core melt frequency,
- o Vessel failure probability given core melt, and
- o Containment failure probability given core melt and vessel failure.

The probability of reactor vessel failure given core melt will, conservatively be taken as 1.0. Containment failure frequency given core melt and vessel failure was determined by the YNPS PSS. From Page 13-46 of that document:

5.27×10^{-2} "best estimate" (taken here as 50% confidence)
2.15×10^{-1} "baseline" (taken here as 95% confidence)

For hazard events greater than 250 mph, containment failure is expected and core melt is assumed. The frequency of release above 250 mph is simply taken as the hazard frequency at 250 mph. At this high speed, wind event frequency is negligible. Tornado frequency at 250 mph is 6×10^{-8} at 50% confidence and 4×10^{-7} at 95% hazard confidence.

Resulting annual release frequencies are as follows:

	Release Frequency < 250 MPH		Total Annual Release Frequency	
Case	95% Confidence	50% Confidence	95% Confidence	50% Confidence
Base Case	1.03×10^{-5}	6.85×10^{-7}	1.07×10^{-5}	7.45×10^{-7}
Cable Tray House Upgrade	2.37×10^{-6}	8.96×10^{-8}	2.77×10^{-6}	1.50×10^{-7}
SSS Upgrade	8.82×10^{-6}	6.32×10^{-7}	9.22×10^{-6}	6.92×10^{-7}
Combined Upgrade	8.17×10^{-7}	3.85×10^{-8}	1.22×10^{-6}	9.85×10^{-8}

The consequences of a release are discussed in the next section. Section 8 combines the above release frequencies with the release consequences in order to assess plant risk.

TABLE 6-1

Data to Assess Relief Valve Challenge Induced LOCA

Event	Mean Value	Reference
PORV FTO	3.75-3	YNPS PSS
	4.27-3	SB PSA
	3.0-4	IREP PG
	1.0-3	*
PORV FTC	1.25-2	YNPS PSS
	2.50-2	SB PSA
	2.0-2	IREP PG
	2.0-2	*
Safety Valve FTO	4.6-5	YNPS PSS
	3.3-4	SB PSA
	1.0-5	IREP PG
	1.0-4	*
Safety Valve FTC	4.6-4	YNPS PSS
	2.9-3	SB PSA
	1.0-2	IREP PG
	2.0-3	*
MOV FTC	1.25-3	YNPS PSS
	4.30-3	SB PSA
	3.0-3	IREP PG
	3.0-3	*

Legend

FTO = Fails to Open

FTC = Fails to Close

YNPS PSS = Yankee Nuclear Power Station Probabilistic Safety Study

SB PSA = Seabrook Station Probabilistic Safety Assessment

IREP PG = Interim Reliability Evaluation Program Procedures Guide (NUREG/CR-2728)

* = Value used in this study

TABLE 6-2

Logic Expression Basic Events

Event	Failure Probability
ASD12RF	6.00×10^{-3}
ASD34RF	6.00×10^{-3}
BATCGR1R	2.94×10^{-4}
BATCGR3R	2.94×10^{-4}
BAT1RUN	1.0
BAT2RUN	1.0
BS5-2R	8.88×10^{-6}
BS6-3R	8.88×10^{-6}
DCBS1R	8.88×10^{-6}
DCBS2R	8.88×10^{-6}
DC1D	3.61×10^{-3}
DC1R	3.00×10^{-3}
DC2R	3.00×10^{-3}
DC3D	3.61×10^{-3}
DC3R	3.00×10^{-3}
DG1R	0.1
DG2R	0.1
DG3R	0.1
EBFP1R	1.0×10^{-2}
EBFP2R	1.0×10^{-2}
EBS1R	8.88×10^{-6}
EBS3R	8.88×10^{-6}
INST1	1.9×10^{-4}

TABLE 6-2 (continued)

Logic Expression Basic Events

Event	Failure Probability
INST2	2.55×10^{-4}
LCLI	0.1
MCC1BS1R	8.88×10^{-6}
MCC2BS2R	8.88×10^{-6}
OERCMOV	0.1
RCMOV1	3.0×10^{-3}
RCMOV2	3.0×10^{-3}
SEBFP	1.0×10^{-2}
SGP1	2.5×10^{-7}
SIPIPE	1.0×10^{-3}
SIP1R	1.0×10^{-2}
SIP2R	1.0×10^{-2}
SIP3R	1.0×10^{-2}
SSSI	3.15×10^{-4}
SSSR	1.0×10^{-2}
UPS1R	2.94×10^{-4}
UPS2R	2.94×10^{-4}
VB1R	8.88×10^{-6}
VBS2R	8.88×10^{-6}
24V2R	0.5
24V3R	0.5
3CHPP	8.23×10^{-2}

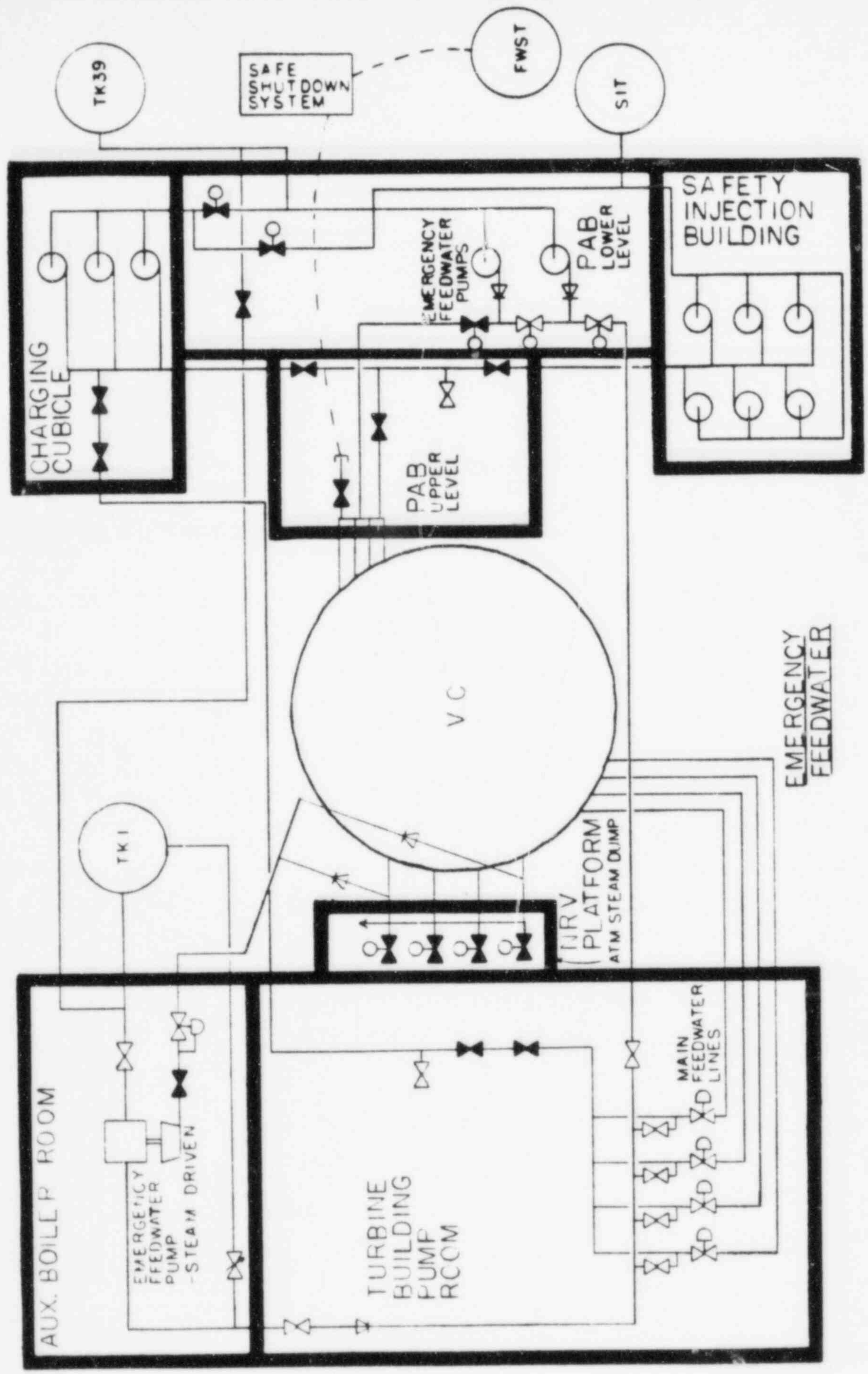


FIGURE 6-1

Feedwater Diagram

7.0 CONSEQUENCE ASSESSMENT

In order to assess the consequences of a release following core melt the characteristics of the release (i.e., its energy and fission product inventory) as well as the plant site characteristics (i.e., weather, population) must be considered. Both aspects have been investigated at length in the YNPS PSS (Reference 3). Sections 7.7.1 and 7.7.2 briefly discuss the various release types and their applicability to this analysis, respectively. Section 7.7.3 discusses consequence results on a per release basis.

7.7.1 YNPS PSS Release Category Discussion

An event tree was developed to logically represent the physical processes associated with core melt and containment of radionuclides, which incorporated pertinent design features for YNPS. Using MARCH and CORCON-MOD1 computer códes, and by manually calculating the process of particle-bed cooling and reactor cavity response, frequencies and timing of releases of radionuclides from the core to within the vapor container were determined. Then, failure modes for containment failures and their frequencies were determined. Further analysis using the CORRAL-2 code established the behavior of radionuclides that escape the vapor container.

Core melt sequences were divided into six types, depending on containment response. These six types are listed below:

1. Large LOCA, in which the reactor vessel pressure is low at the onset of core melt.

2. LOCAs and transients, in which the reactor vessel pressure is high at the onset of core melt and where there is some means for injecting water into the lower plenum.

3. TMLB' and ATWS sequences for which there is no means for injecting water into the lower plenum or when the power level is high enough that any water which is available is not effective in cooling the core debris in the lower plenum.

4. Steam generator tube ruptures (SGTRs) in which the primary system is not isolated from the affected steam generator.

5. Non-isolable LOCAs outside the vapor container.

6. Reactor vessel ruptures.

Each group was handled differently when quantifying the release frequencies.

Based on MARCH and CORRAL-2-EI results, six release categories were identified that cover the spectrum of release magnitudes and timings possible from severe accident sequences. A discussion of these release categories follows:

- <u>Release Categories 1 (RC1) and 1A (RC1A)</u> cover core melt accidents in which the primary vessel lower head fails, molten core mixes violently with the water in the neutron shield tank (a steam explosion is assumed to occur), and major structural failure of the containment occurs, resulting in a rapid release of radioactivity into the environment. RC1 covers all of the accidents in which this type of failure occurs two hours or later into the accident. RC1A covers accidents in which the failure occurs approximately one hour into the accident. A high rate of energy release from the containment would be expected for these categories. Also, after the structural failure of the containment, any additional fission product release from the core would be released directly into the atmosphere. RC1 and RC1A result in the highest release fractions of all six categories because of the short time period between core melt and containment rupture, and because of the manner of containment rupture.

  RC1 corresponds roughly with WASH-1400's PWR1 release category. Both involve an early energetic containment rupture and both involve an additional oxidation (or steam explosion) release component. However, the PWR1 category involves a postulated

catastrophic steam explosion within the primary system, while for the YNPS such an explosion within the primary system is not considered to be possible. The RC1 category containment failure results from a possible steam explosion once the reactor vessel head has failed and the molten core mixes with the water contained in the neutron shield tank.

- Release Category 2 (RC2) covers core melt accidents in which containment failure occurs soon after the reactor vessel head fails. This category covers the containment failure modes of early overpressurization and early hydrogen-burn-induced overpressurization. RC2 corresponds most closely to the PWR2 category in WASH-1400 Reactor Safety Study (RSS).

- Release Category 3 (RC3) includes three distinct containment failure modes. The first is overpressurization due to hydrogen combustion failure. This failure mode is assumed to occur an hour or more after primary head failure and is distinct from the overpressurization failure immediately after head failure which is included in RC2. The second type of containment failure mode is failure of the reactor cavity concrete after only partial melt-through has occurred. (It is assumed that the containment shell fails immediately after the cavity fails.) Finally, the large leakage core melt accidents are also included in this release category. Examination of the containment failure modes for RC3 indicates that the hydrogen burn failure is clearly dominant. Therefore, the release characteristics for this category are assumed to be those associated with hydrogen-burn-induced overpressurization.

  Release Category 3 does not correspond directly with any of the RSS release categories.

- Release Category 4 (RC4) represents core melt accidents in which containment failure is not expected to occur until the containment floor has almost entirely melted through. In such cases, the

failure would not occur until many hours after the core has melted. The energy release would be much lower than those for RC1, RC2, and RC3.

Release Category 4 does not correspond directly to any WASH-1400 categories. The reason for this is that melt-through-induced containment failure at the YNPS results in release directly into the atmosphere because of the elevated containment sphere. Melt-through failures for the Surry plant analyzed in WASH-1400 result in release to the ground, where significant fission product retention would occur.

- Release Category 5 (RC5) represents accident sequences in which core melt occurs, but where containment integrity is preserved. Release occurs due to normal allowable leakage from the containment. The release fractions for RC5 are much smaller than those for any of the other release categories.

The philosophy followed in the YNPS PSS to predict fission product release was to use the Reactor Safety Study (RSS) methodology and then to adjust the results in order to conduct a probability of frequency evaluation. An overview of the methodology is described below.

Fission products' transport and deposition within the containment and their release to the atmosphere were predicted with the CORRAL-2 computer code. Results of these predictions were used to group the various accident sequences and containment failure modes into representative release categories based on release amounts, time and duration of releases, and release energies. At this point, the release results were adjusted to account for conservatisms which lie mainly in the core source terms for the various fission product species. The representative release categories were input to the CRAC2 computer code to determine consequences. (CORRAL-2 direct output represents an upper-bound result, called 95% confidence level. The median estimate was based on CORRAL-2 results reduced by approximately a factor of 3, and the lower bound called 5% confidence level, was based on CORRAL-2 results reduced by a factor of approximately 9.)

Characteristics of the release categories are provided in Tables 7-1 and 7-2.

7.7.2 Use of the YNPS PSS Release Category Information

Section 10.9 of Reference 3 discusses the grouping of ıeleases due to various core melt event sequences into appropriate release categories. As shown in Section 10.9, for each initiating event type, the total annual release frequency can be divided into release categories by percent of total release. For all the initiating events of interest in this analysis, for hazard intensities up to 250 MPH, the annual release frequency breaks down as follows (from table 13-6 of Reference 3):

- RC1 15.2%
- RC2 17.1%
- RC3 60.5%
- RC4 7.2%

Note that RC5, containment leakage, is assumed to always be present in addition to the above.

For events involving hazard intensities above 250 MPH, direct containment failure and early core melt are assumed. Release category RC1A is considered appropriate for this case.

These release category distributions are used to calculate release consequences and evaluate risk.

7.7.3 Results

In the YNPS PSS, consequences of the various release categories were analyzed using the computer code CRAC2. Site-specific parameters - meteorology, topography, and evacuation - were used to produce the plant risk profile.

CRAC2 output was examined and individual, societal, and person-rem risk levels were determined for each release category assuming the following evacuation assumptions:

1. No evacuation during or following a hurricane. Sheltering was assumed.

2. For tornados, a 3-hour delay followed by evacuation at 10 mph was used.

The bases for these assumptions are provided below. The conditional per release values for each of these risk indices are provided in Table 7-3.

In the hurricane situation, people are assumed to take shelter in the building structure. The hurricane could last for 5 to 10 hours, including 3 to 6 hours of intensive activity, and could affect the entire emergency planning zone. Heavy rain and high wind could cause flooding and road damage and thus preclude evacuation.

In the tornado situation, people are assumed to evacuate with a speed of 10 mph and a delay time of 3 hours. A tornado affects a small area and lasts for a short time. However, the accompanying thunderstorm could cause some delay to the evacuation.

The difference in acute fatality risk is caused by the difference in evacuation assumptions. However, latent cancer fatality and population dose are insensitive with respect to the evacuation assumption and thus remain the same.

Based on the conditional frequencies assigned to each release category in Section 7.7.2, the following averages for each risk index (per release) were developed from the CRAC2 results in Table 7-3.

Containment Integrity Not Impacted by Hazard

Event	Individual Acute	Societal Latent Cancer Per Person	Person-Rem
Tornado	4.7×10^{-5}	2.4×10^{-5}	6.6×10^{5}
Hurricane	6.0×10^{-4}	2.4×10^{-5}	6.6×10^{5}

Containment Integrity Impacted by Hazard

Event	Individual Acute	Societal Latent Cancer Per Person	Person-Rem
Tornado	6.0×10^{-4}	5.5×10^{-5}	1.3×10^{6}
Hurricane	2.5×10^{-3}	5.5×10^{-5}	1.3×10^{6}

To conservatively envelope individual acute fatality risk estimates (Section 8), the hurricane values will be used for hazard intensities not impacting containment integrity. Note that this could result in a factor of 4 to 13 conservatism for sequences in which tornado-induced damage dominates. For hazard intensities impacting containment integrity (>250 mph), tornado values will be used since the frequency of high wind/hurricane type events exceeding 250 mph is much less than tornado events greater than 250 mph.

TABLE 7-1

Associated Release Category Parameters Required for Consequence Calculations

Release Category	Time of Release[1] (hr)	Duration of Release (hr)	Elevation of Release (m)	Energy Release (10^6 Btu/hr)	Evacuation Warning Time[2] (hr)
RC1	2.0	0.5	0.0	80.0	2.0
RC1A	1.0	0.5	0.0	80.0	1.0
RC2	1.0	0.5	0.0	80.0	1.0
RC3	2.5	0.5	0.0	60.0	2.5
RC4	50.0	3.0	0.0	0.2	50.0
RC5	1.0	10.0	0.0	0.0005	1.0

1 Time from initiation of accident to start of significant release.

2 Time from point at which it is known that significant release might occur to the start of significant release.

TABLE 7-2

Release Fractions - 5% Bound, 50% Confidence Level and 95% Bound

Release Category		Kr-Xe[1]	OI[2]	I_2[2]	Cs-Rb[2]	Te[3]	Ba-Sr[2]	Ru[2]	La[3]
RC1	95%	0.99	0.0070	0.73	0.67	0.77	0.17	0.44	0.012
and	50%	0.99	0.0023	0.24	0.22	0.51	0.06	0.15	0.008
RC1A	5%	0.99	0.0007	0.07	0.07	0.15	0.02	0.04	0.002
RC2	95%	0.96	0.0068	0.62	0.49	0.37	0.14		0.0032
	50%	0.96	0.0023	0.21	0.16	0.25	0.05		0.0021
	5%	0.96	0.0007	0.06	0.05	0.07	0.01	0.004	0.0006
RC3	95%	1.00	0.0070	0.15	0.14	0.18	0.033		0.0030
	50%	1.00	0.0023	0.05	0.05	0.12	0.011		0.0020
	5%	1.00	0.0007	0.02	0.01	0.04	0.003	0.003	0.0006
RC4	95%	1.00	0.0070	0.0098	$4.0(10)^{-4}$	$4.0(10)^{-4}$	$1.0(10)^{-4}$		$5.3(10)^{-6}$
	50%	1.00	0.0023	0.0033	$1.3(10)^{-4}$	$2.7(10)^{-4}$	$0.3(10)^{-4}$		$3.5(10)^{-6}$
	5%	1.00	0.0007	0.0010	$0.4(10)^{-4}$	$0.8(10)^{-4}$	$0.1(10)^{-4}$	$0.6(10)^{-5}$	$1.1(10)^{-6}$
RC5	95%	0.02	0.00010	3.4×10^{-4}	$2.1(10)^{-4}$	$2.2(10)^{-4}$	$5.6(10)^{-5}$		$2.8(10)^{-6}$
	50%	0.02	0.00003	1.1×10^{-4}	$0.7(10)^{-4}$	$1.4(10)^{-4}$	$3.7(10)^{-5}$		$1.9(10)^{-6}$
	5%	0.02	0.00001	0.3×10^{-4}	$0.2(10)^{-4}$	$0.4(10)^{-4}$	$0.6(10)^{-5}$	$0.3(10)^{-5}$	$0.6(10)^{-6}$

1 No reduction from the base case (95% bound) is assumed.

2 Factor of 3 reduction from base case (95% bound) for 50% confidence level, and a factor of 10 from base case for 5% bound. Note - For Ru group, these reductions apply only to RC1 and RC1A.

3 Factor of 1.5 reduction from base case (95% bound) for 50% confidence level, and a factor of 5 from base case for 5% bound. Note - For Ru group, these reductions apply only to RC2, RC3, RC4, and RC5.

TABLE 7-3

Expected Conditional Individual, Societal, and Person-Rem Risk Level Values Per Release

Release Category	Acute Fatality Risk per Person Within 1 Mile of the Plant: Hurricane	Acute Fatality Risk per Person Within 1 Mile of the Plant: Tornado	Latent Cancer Fatality Risk per Person Within 50 Miles	Person-Rem Within 50 Miles
RC1	2.4×10^{-3}	3.1×10^{-4}	5.4×10^{-5}	1.3×10^{6}
RC1A	2.5×10^{-3}	6.0×10^{-4}	5.5×10^{-5}	1.3×10^{6}
RC2	1.4×10^{-3}	0.0	3.3×10^{-5}	1.0×10^{6}
RC3	0.0	0.0	1.6×10^{-5}	4.8×10^{5}
RC4	0.0	0.0	1.0×10^{-7}	3.0×10^{3}
RC5	0.0	0.0	4.2×10^{-8}	1.3×10^{3}

Latent cancer fatality and whole body dose are insensitive to the evacuation assumptions.

8.0 RISK ASSESSMENT

The results of the analysis performed in Sections 6.7 and 7.3 can be combined to develop quantitative estimates of individual, societal, and person-rem risk levels. Table 8-4 provides the results of this evaluation for each of the plant configurations evaluated for the spectrum of wind events from 0 to infinity.

The following tables develop the Individual Acute Fatality Risk, Societal Latent Cancer Fatality Risk per person and Person-Rem Exposure from data developed in Section 6.0 and 7.0 for each plant configuration considered as Tables 8-1, 8-2, and 8-3, respectively.

Table 8-5 provides a comparison of the NRC preliminary safety goals (Reference 1) to risks resulting from each plant configuration analyzed. Note that the "Base Case" results in individual and societal risk levels less than those discussed in Reference 1.

TABLE 8-1

Individual Acute Fatality Risk Development

Case Description		Core Melt and Release Frequency(yr^{-1}) <250 MPH	Individual Acute Fatality Risk <250 MPH	Individual Acute Fatality Risk (yr^{-1}) <250 MPH	Core Melt and Release Frequency (yr^{-1}) >250 MPH	Individual Acute Fatality Risk >250 MPH	Individual Acute Fatality Risk (yr^{-1}) > 250 MPH	Total Individual Acute Fatality Risk (yr^{-1})
Base Case	50	6.85-7	6.0-4	4.11-10	6.0-8	6.0-4	3.6-11	4.47-10
	95	1.03-5	6.0-4	6.2-9	4.0-7	6.0-4	2.4-10	6.42-9
Cable Tray House Upgrade	50	8.96-8	6.0-4	5.38-11	6.0-8	6.0-4	3.6-11	8.98-11
	95	2.37-6	6.0-4	1.42-9	4.0-7	6.0-4	2.4-10	1.66-9
SSS Upgrade	50	6.32-7	6.0-4	3.8-10	6.0-8	6.0-4	3.6-11	4.15-10
	95	8.82-6	6.0-4	5.3-9	4.0-7	6.0-4	2.4-10	5.53-9
Combined Upgrade	50	3.85-8	6.0-4	2.31-11	6.0-8	6.0-4	3.6-11	5.91-11
	95	8.17-7	6.0-4	4.9-10	4.0-7	6.0-4	2.4-10	7.3-10

TABLE 8-2

Societal Latent Cancer Fatality Risk Development Per Person

Case Description		Core Melt and Release Frequency(yr^{-1}) < 250 MPH	Conditional Societal Latent Cancer Fatality Risk <250 MPH	Societal Latent Cancer Fatality Risk (yr^{-1}) <250 MPH	Core Melt and Release Frequency (yr^{-1}) > 250 MPH	Conditional Societal Latent Cancer Fatality Risk >250 MPH	Societal Latent Cancer Fatality Risk (yr^{-1}) >250 MPH	Total Societal Latent Cancer Fatality Risk (yr^{-1})
Base Case	50	6.85-7	2.4-5	1.64-11	6-8	5.5-5	3.3-12	1.97-11
	95	1.03-5	2.4-5	2.48-10	4-7	5.5-5	2.2-11	2.7-10
Cable Tray House Upgrade	50	8.96-8	2.4-5	2.15-12	6-8	5.5-5	3.3-12	5.45-12
	95	2.36-6	2.4-5	5.68-11	4-7	5.5-5	2.2-11	7.88-11
SSS Upgrade	50	6.32-7	2.4-5	1.52-11	6-8	5.5-5	3.3-12	1.85-11
	95	8.8-6	2.4-5	2.12-10	4-7	5.5-5	2.2-11	2.34-10
Combined Upgrade	50	3.85-8	2.4-5	9.23-13	6-8	5.5-5	3.3-12	4.22-12
	95	8.17-7	2.4-5	1.96-11	4-7	5.5-5	2.2-11	4.16-11

TABLE 8-3

Person-Rem Exposure Development

Case Description		Core Melt and Release Frequency(yr^{-1}) < 250 MPH	Conditional Person-Rem Exposure <250 MPH	Person-Rem Exposure (yr^{-1}) < 250 MPH	Core Melt and Release Frequency (yr^{1}) >250 MPH	Conditional Person-Rem Exposure >250 MPH	Person-Rem Exposure (yr^{-1}) >250 MPH	Total Person-Rem Exposure (yr^{-1})
Base Case	50	6.85-7	6.6+5	0.45	6-8	1.3+6	7.8-2	0.53
	95	1.03-5	6.6+5	6.81	4-7	1.3+6	0.52	7.33
Cable Tray House Upgrade	50	8.96-8	6.6+5	5.91-2	6-8	1.3+6	7.8-2	0.14
	95	2.37-6	6.6+5	1.56	4-7	1.3+6	0.52	2.08
SSS Upgrade	50	6.32-7	6.6+5	0.42	6-8	1.3+6	7.8-2	0.50
	95	8.8-6	6.6+5	5.82	4-7	1.3+6	0.52	6.34
Combined Upgrade	50	3.85-8	6.6+5	2.54-2	6-8	1.3+6	7.8-2	0.10
	95	8.17-7	6.6+5	5.4-1	4-7	1.3+6	0.52	1.06

TABLE 8-4

Risk Levels

Description	Hazard Curve	Individual Acute Fatality Risk	Societal Latent Cancer Risk (Per Person)	Person-Rem Exposure
Base Case	50%	4.5×10^{-10}	2.0×10^{-11}	0.53
	95%	6.4×10^{-9}	2.7×10^{-10}	7.33
Cable Tray	50%	9.0×10^{-11}	5.45×10^{-12}	0.14
House Upgrade	95%	1.7×10^{-9}	7.9×10^{-11}	2.08
SSS Upgrade	50%	4.15×10^{-10}	1.85×10^{-11}	0.50
	95%	5.5×10^{-9}	2.34×10^{-10}	6.34
Combined Upgrade	50%	5.9×10^{-11}	4.22×10^{-12}	0.10
	95%	7.3×10^{-10}	4.16×10^{-11}	1.06

TABLE 8-5

Safety Goal Comparison

Description	Hazard Curve	% of Individual Acute Fatality Risk Goal	% of Societal Risk Goal	% of Core Melt Goal
Base Case	50%	0.09%	0.001%	13%
	95%	1.0%	0.01%	48%
Cable Tray	50%	0.02%	0.0003%	2.0%
House Upgrade	95%	0.34%	0.004%	11%
SSS Upgrade	50%	0.08%	0.001%	12%
	95%	1.0%	0.01%	41%
Combined Upgrade	50%	0.01%	0.0002%	0.73%
	95%	0.15%	0.002%	4.0%

9.0 COST-BENEFIT ASSESSMENT

As shown in Section 8.0, both individual and societal risk safety goals, and the core melt frequency design objective are met for the base case plant configuration including the SSS designed for seismic. Since this system will be installed, it is the appropriate plant configuration from which to assess the benefit of additional design modifications. Table 9-1 provides the results of the investigation of the two backfits discussed in Section 1.0.

Neither backfit is justified from a cost-benefit perspective for the reasons provided in Section 1.0.

9.1 Costs for Structural Upgrade

Cost estimates have been developed and are summarized in this section for proposed modifications to structures/components. The costs presented for the proposed 10^{-4} event and 10^{-5} event structural modifications include both direct and indirect dollars. All aspects of the modifications are accounted for including equipment, labor, management, engineering, etc. Yankee seismic structural modifications control design of structures for windspeed through the 10^{-4} event.

The estimated modification costs associated with the events are as follows:

Event	Cost
10^{-4}	$108,000*
10^{-5}	$296,000**

*The 10^{-4} event is governed by the 110 mph straight wind and modification of the Cable Tray House.

**Structures/components requiring modifications for the 10^{-5} event are the Turbine Building SW Stairwell, the PAB North Wall, the NRV Enclosure, the Fire Pumphouse Enclosure, Fire Water Storage Tank and the SSS Pumphouse.

TABLE 9-1

Cost-Benefit Analysis Results

Description	(1) Plant Capacity (mph)	SSS DSN Capacity (mph)	SSS Actual Capacity (mph)	Hazard Curve (%)	Core Melt Freq. Per Year	Indiv. Risk Per Year	Societal Risk Per Year	Residual Person-Rem Per Year	Reduction in Person-Rem Per Year	(3) Just. Costs to Upgrade ($'s)	Actual Costs of Upgrade ($'s)	Ratio of Actual to Justif. Costs
Base Case	70(2)	--	175	50	1.3-5	4.47-10	1.97-11	0.53	--	--	-	-
				95	4.8-5	6.43-9	2.7-10	7.33	--	--	-	-
Cable Tray House Upgrade	160	--	175	50	1.8-6	8.98-11	5.45-12	0.14	0.39	3.9K	108K	28
				95	1.1-5	1.66-9	7.88-11	2.08	5.25	52.5K	108K	2
SSS Upgrade	70(2)	165	250	50	1.2-5	4.15-10	1.85-11	0.50	0.03	0.3K	296K	987
				95	4.1-5	5.53-9	2.34-10	6.34	0.99	9.9K	296K	30
Combined Upgrade	160	165	250	50	7.9-7	5.91-11	4.22-12	0.1	0.43	4.3K	404K	94
				95	4.2-6	7.3-10	4.16-11	1.06	6.27	62.7K	404K	6
Combined Upgrade Compared to Cable Tray House Upgrade	160	165	250	50	7.3-7	5.91-11	4.22-12	0.1	0.03	0.3K	296K	987
				95	4.2-6	7.3-10	4.16-11	1.06	1.02	10.2K	296K	29

(1) Excluding Safe Shutdown System.

(2) The Cable Tray House fails at ≃ 70 mph, this analysis assumes the Cable Tray House failure fails <u>all</u> normal plant instrumentation yielding a core melt probability of 10^{-1} above 70 mph since only local instrumentation is credited. This is extremely conservative for reasons stated in the analysis.

(3) Based on $1,000 per person-rem averted for 10 years or $10,000/person-rem.

10.0 REFERENCES

1. 48FR10772, March 14, 1983, "Safety Goal Development".

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

3. YAEC Letter to USNRC, dated January 3, 1983 (FYR 83-1).

4. EPRI-NP-801, "ATWS: A Re-Appraisal Part III - Frequency of Anticipated Transients".

5. McDonald, J. R., 1980, "Tornado and Straight Wind Hazard Probability for Yankee Rowe Nuclear Power Plant Site", prepared for USNRC.

6. Owners and Engineering Firms Informal Group on Concrete Masonry Walls, "Reassessment of Safety-Related Concrete Masonry Walls", October 6, 1980.

7. American concrete Institute, "Building Code Requirements for Concrete Masonry Structures", ACI 531-79 (Revised 1981).

8. American National Standard, "Minimum Design Loads for Buildings and Other Structures", ANSI A58.1 - 1982.

9. Simiu, E., and R. H. Scanlan, Wind Effects on Structures: An Introduction to Wind Engineering, J. Wiley & Sons, 1978.

10. Twisdale, L. A., "Wind Loading Frequencies and Transmission Line Design", paper presented at Southeastern Electric Exchange., 1984 Conference, Bal Harbour, Florida, April 1984.

11. Twisdale, L. A., "Wind Loading Underestimate in Transmission Line Design", Transmission Distribution, December 1982.

12. CYGNA Energy Services, "Preliminary Review of Masonary Walls at Yankee Nuclear Power Station at Rowe", October 1981. Appendix A, Wall location drawings.

13. NUREG-0825, June 1983, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Yankee Nuclear Power Station, Final Report".

APPENDIX A

Acronym Table

BR	AUX BOILER ROOM HAZARD/LOCATION FAILURE
SD12R(F)	ATM STEAM DUMP VALVE 1 OR 2 FAILS TO OPEN - RANDOM
SD34R(F)	ATM STEAM DUMP VALVE 3 OR 4 FAILS TO OPEN - RANDOM
	INITIATING EVENT - LOSS OF OFFSITE POWER
ATCGR1R	RANDOM FAILURE OF BATTERY CHARGER NUMBER 1
ATCGR2R	RANDOM FAILURE OF BATTERY CHARGER NUMBER 2
AT1RUN	BATTERY NUMBER 1 FAILURE TO RUN FOR 24 HOURS
AT2RUN	BATTERY NUMBER 2 FAILURE TO RUN FOR 24 HOURS
S5-2R	480V BUS 5-2 RANDOM FAILURE
S6-3R	480V BUS 6-3 RANDOM FAILURE
	TOP EVENT - ATMOSPHERIC STEAM DUMP AVAILABLE
BLTPV	CABLE TRAY HOUSE (PORV POWER) HAZARD/LOCATION FAILURE
BLT	CABLE TRAY HOUSE (EXCEPT PORV POWER) HAZARD/LOC FAILURE
HBLDN	FLOW PATH - CHARGING PUMPS TO BLOWDOWN HEADER
HMFWH	FLOW PATH - CHARGING PUMPS TO MAIN FEED HEADER
	TOP EVENT - OPERATOR ACTUATES ATM STEAM DUMP
CBS1R	RANDOM FAILURE OF DC BUS NUMBER 1
CBS2R	RANDOM FAILURE OF DC BUS NUMBER 2
C1D	DEMAND FAILURE OF BATTERY NUMBER 1
C1L	BATTERY ROOM NUMBER 1 HAZARD/LOCATION FAILURE
C1RUN	FAILURE OF DC POWER SUPPLY NUMBER 1
C1R	125V DC BUS NUMBER ONE FAILS TO REMAIN ENERGIZED
C2L	BATTERY ROOM NUMBER 2 HAZARD/LOCATION FAILURE
C2RUN	FAILURE OF DC POWER SUPPLY NUMBER 2
C2R	125V DC BUS NUMBER TWO FAILS TO REMAIN ENERGIZED
C3D	DEMAND FAILURE OF BATTERY NUMBER 3
C3R	125V DC BUS NUMBER THREE FAILS TO REMAIN ENERGIZED
DG1L	DIESEL GENERATOR 1 HAZARD/LOCATION FAILURE
DG1R	DIESEL GENERATOR NUMBER 1 FAILS TO SUPPLY POWER
DG2L	DIESEL GENERATOR 2 HAZARD/LOCATION FAILURE
DG2R	DIESEL GENERATOR NUMBER 2 FAILS TO SUPPLY POWER
DG3L	DIESEL GENERATOR 3 HAZARD/LOCATION FAILURE
DG3R	DIESEL GENERATOR NUMBER 3 FAILS TO SUPPLY POWER
D	TOP EVENT - POWER AVAILABLE TO PORV
EBFBLDN	FLOW PATH - ELEC EM FEED PPS TO BLOWDOWN HEADER
EBFMFWH	FLOW PATH - ELEC EM FEED PPS TO MAIN FEED HEADER
EBFP1R	ELECTRIC EMERGENCY BOILER FEEDWATER PUMP 1
EBFP2R	ELECTRIC EMERGENCY BOILER FEEDWATER PUMP 2
EBS1R	RANDOM FAILURE OF EMERGENCY 480V BUS NUMBER 1
EBS3R	RANDOM FAILURE OF EMERGENCY 480V BUS NUMBER 3
E	TOP EVENT - POWER OPERATED RELIEF VALVE OPENS
FWST	FIRE WATER STORAGE TANK HAZARD/LOCATION FAILURE
F	TOP EVENT - POWER OPERATED RELIEF VALVE RECLOSES
G	TOP EVENT - POWER AVAILABLE TO PORV BLOCK VALVE
H	TOP EVENT - PORV BLOCK VALVE CLOSES
INSA	MAIN CONTROL ROOM INSTRUMENTATION
INST1	RANDOM FAILURE OF INSTRUMENTATION FOR CHANNEL 1
INST2	RANDOM FAILURE OF INSTRUMENTATION FOR CHANNEL 2
INS1	INSTRUMENTATION CHANNEL NUMBER 1
INS2	INSTRUMENTATION CHANNEL NUMBER 2
INS	INSTRUMENTATION FAILURE
I	TOP EVENT - 1 OF 2 PRIMARY CODE SAFETY VALVES OPENS
J	TOP EVENT - PRIMARY CODE SAFETY VALVE(S) RECLOSE
K	TOP EVENT - MCS LOOP SAFETY VALVE(S) OPEN
LA	INITIATING EVENT - LOCA

B	TOP EVENT - SCRAM
CLI	LOCAL INSTRUMENTATION
C	TOP EVENT - POSITIVE REACTIVITY CONTROL
D	TOP EVENT - SAFETY INJECTION
E	TOP EVENT - RECIRCULATION
F	TOP EVENT - FEEDWATER ADDITION AND CONTROL
G	TOP EVENT - STEAM REMOVAL AND CONTROL
H	TOP EVENT - INSTRU AVAIL SUFFICIENT TO CONTROL PLANT
LPAB	LOWER LEVEL PRIMARY AUX BUILDING HAZARD/LOC FAILURE
	TOP EVENT - MCS LOOP SAFETY VALVE(S) RECLOSE
CC1BS1R	MOTOR CONTROL CENTER NUMBER 1 BUS NUMBER 1 RANDOM FAILURE
CC2BS2R	MOTOR CONTROL CENTER NUMBER 2 BUS NUMBER 2 RANDOM FAILURE
CR	MAIN CONTROL ROOM HAZARD/LOCATION FAILURE
RVL	NON-RETURN VALVE PLATFORM HAZARD/LOCATION FAILURE
A	INITIATING EVENT - LOSS OF OFFSITE POWER
B	TOP EVENT - MAIN COOLANT SYSTEM IS ISOLATED
C	TOP EVENT - SCRAM
D	TOP EVENT - POSITVE REACTIVITY CONTROL (RE STM REMOVAL)
ERCMOV	OPERATOR ERROR - FAILS TO OPEN RECIRC VALVE MANUALLY
E	TOP EVENT - FEEDWATER ADDITION AND CONTROL
F	TOP EVENT - STEAM REMOVAL AND CONTROL
G	TOP EVENT - INSTRU AVAIL SUFFICIENT TO CONTROL PLANT
PPRM	TURBINE BUILDING PUMP ROOM HAZARD/LOCATION FAILURE
PWRCH1	INSTRUMENT CHANNEL NUMBER 1 POWER SUPPLY
PWRCH2	INSTRUMENT CHANNEL NUMBER 2 POWER SUPPLY
RCMOV1	MOTOR OPERATED RECIRCULATION VALVE 1
RCMOV2	MOTOR OPERATED RECICRULATION VALVE 2
SEBFMFWH	FLOW PATH - STEAM EM FEED TO MAIN FEED HEADER
SEBFP	STEAM DRIVEN EMERGENCY BOILER FEEDWATER PUMP
SGPI	STEAM GENERATOR PRESSURE INSTRUMENTATION
SIBLDN	FLOW PATH - SAFETY INJECTION TO BLOWDOWN HEADER
SIBN	SAFETY INJ BUILDING NORTH WALL HAZARD/LOC FAILURE
SIB	SAFETY INJECTION BUILDING HAZARD/LOCATION FAILURE
SIPIPE	ALL SAFETY INJECTION FLOW PATH COMPONENTS
SIP1R	SAFETY INJECTION PUMP 1 - RANDOM FAILURE
SIP2R	SAFETY INJECTION PUMP 2 - RANDOM FAILURE
SIP3R	SAFETY INJECTION PUMP 3 - RANDOM FAILURE
SIT	SAFETY INJECTION TANK HAZARD/LOCATION FAILURE
SI1R	SAFETY INJECTION TRAIN NUMBER 1 TO THE SECONDARY
SI2R	SAFETY INJECTION TRAIN NUMBER 2 TO THE SECONDARY
SI3R	SAFETY INJECTION TRAIN NUMBER 3 TO THE SECONDARY
SSSBLDN	FLOW PATH - SAFE SHUTDOWN SYS TO BLOWDOWN HEADER
SSSINST	SAFE SHUTDOWN SYSTEM INSTRUMENTATION RANDOM FAILURE
SSSI	SAFE SHUTDOWN SYSTEM INSTRUMENTATION
SSSL	SAFE SHUTDOWN SYSTEM HAZARD/LOCTION FAILURE
SSSR	SAFE SHUTDOWN SYSTEM - RANDOM FAILURE
SWGR	SWITCHGEAR ROOM HAZARD/LOCATION FAILURE
TK1L	WATER SOURCE - TANK 1 HAZARD/LOCATION FAILURE
TK1	WATER SOURCE - TANK NUMBER ONE
TK39	WATER SOURCE - TANK 39 HAZARD/LOCATION FAILURE
ULPAB	UPPER LEVEL PRIMARY AUX BUILDING HAZARD/LOC FAILURE
UPS1R	UNINTERRUPTIBLE POWER SUPPLY NO. 1 RANDOM FAILURE
UPS2R	UNINTERRUPTIBLE POWER SUPPLY NO. 2 RANDOM FAILURE
VBS2R	RANDOM FAILURE OF VITAL BUS NUMBER 2
VB1R	RANDOM FAILURE OF VITAL BUS NUMBER 1

V2R	2400V BUS NUMBER 2 RANDOM FAILURE
V3R	2400V BUS NUMBER 3 RANDOM FAILURE
HPP	3 CHARGING PUMPS - FAILURE IS ANY ONE OF THREE FAILS