MONTICELLO
INDIVIDUAL PLANT EXAMINATION
OF EXTERNAL EVENTS (IPEEE)

NSPLMI-95001

Revision 1

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EXECUTIVE SUMMARY

This report documents Northern States Power Company's (NSP) response to Supplement 4 of Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," which was issued in June of 1991. The IPEEE extends the analysis performed for the individual plant examination of internal events (IPE) which is the subject of the generic letter and its first three supplements. NSP's IPE report for the Monticello plant was submitted to the NRC for review on February 27, 1992. The NRC transmitted its evaluation of the report to NSP on May 26, 1994, concluding that "...the staff finds the licensee's IPE conclusion that no fundamental weakness or severe accident vulnerabilities exist at Monticello to be reasonable."

The IPEEE assessments described in this report address the external events identified in Supplement 4 of Generic Letter 88-20, namely internal fires, high winds, floods, and other credible external events. The analysis of seismic events was conducted using a seismic margins assessment in accordance with the guidance provided in Supplement 5 to Generic Letter 88-20. For the internal fire assessment, a probabilistic risk assessment approach was used, using an updated version of the IPE as a basis, combined with the deterministic evaluation techniques of EPRI's fire-induced vulnerabilities evaluation (FIVE) methodology. The evaluation of other external events was done by comparing the plant to the NRC's 1975 Standard Review Plan. The analyses for these assessments began in 1992.

The primary objectives of the IPEEE, as stated by the NRC in the generic letter, are for each utility to develop an appreciation of severe accident behavior; understand the most likely severe accident sequences that could occur at its plant under full-power operating conditions; gain a qualitative understanding of the overall likelihood of core damage and radioactive material releases; and, if necessary, reduce the overall likelihood of core damage and radioactive releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

By letter dated January 5, 1995, Monticello notified the staff of a change in the manner in which the seismic portion of the IPEEE will be completed. This change is based on new information regarding large reductions in the seismic hazard estimates for sites in the eastern United States, as presented in draft NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," issued in April of 1994. As stated in the January 5th letter, the assessment of the seismic IPEEE was to be completed in conjunction with the schedule for submitting information that addresses Unresolved Safety Issue (USI)A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants." The fundamental result of the seismic margins assessment is that the majority of components included in the assessment were found to have relatively high seismic capacity. Those components not meeting the screening criteria were either shown to be adequate at the safe shutdown earthquake (SSE) level, or are to be upgraded under Monticello's Seismic Qualification Utilities Group (SQUG) effort associated with USI A-46. Based on these findings, it was concluded that there are no significant vulnerabilities to severe accidents that exist at Monticello that would be attributable to seismic events.

The principal finding of the fire portion of the IPEEE is that there is no area in the plant in which a fire would lead directly to the inability to cool the core. Without additional random equipment failures unrelated to damage caused by the fire, core damage will not occur. As a result, this study concludes that there are no vulnerabilities due to fire events at the Monticello Nuclear Generating Station. The calculated core damage frequency due to internal fires is not dominant with respect to overall plant risk. With the removal of some of the conservatisms in the analysis, the already low core damage frequency would be reduced even further.

No other external event, such as high winds, floods, or transportation-related accidents, was found to be a safety concern to the plant. No vulnerability was identified, and the screening criteria in NUREG-1407 and Generic Letter 88-20, Supplement 4, are satisfied for all the "other" events suggested in the generic letter and NUREG-1407.

Because no vulnerabilities to external events (seismic, internal fires, or "other") were identified during this assessment, no change to plant hardware or procedures is recommended. This report completes commitments made in regard to the generic letter with respect to the IPEEE.

1. EXAMINATION DESCRIPTION

1.1 Introduction

In July and August of 1985, the NRC published its policy statement on issues related to severe accidents in NUREG-1070. The severe accident policy states that on the basis of currently available information, existing plants pose no undue risk to the health and safety of the public. Therefore, the NRC sees no justification to take immediate action on generic rule-making or other regulatory changes for existing plants because of issues related to severe accidents. The Commission's conclusion of no undue risk is based upon actions taken as a result of the Three Mile Island action plan (NUREG-0737), information from research sponsored by the NRC and the nuclear industry, information from published probabilistic risk assessments and operating experience, and the results of the Industry Degraded Core Rulemaking (IDCOR) technical program.

After November of 1988, the NRC staff issued Generic Letter 88-20 and three supplements which formalized the requirement for an individual plant examination (IPE) under 10 CFR 50.54(f). This generic letter required utilities to perform the IPE; supplement 1 provided reporting requirements; supplement 2 identified accident management strategies to be considered as part of the IPE; and supplement 3 established containment performance improvement considerations. The generic letter and its first three supplements were addressed in the Monticello IPE submittal to the NRC on February 27, 1992. The staff evaluation of the Monticello IPE was received on May 26, 1994, and concluded that NSP met the intent of Generic Letter 88-20 for Monticello.

Supplement 4 to Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," was issued in June of 1991. The IPEEE is to specifically address seismic and internal fire events, and other external events such as high winds and floods.

The primary objectives of the IPEEE, as stated by the NRC in the generic letter, are for each utility to develop an appreciation of severe accident behavior; understand the most likely severe accident sequences that could occur at its plant under full-power operating conditions; gain a qualitative understanding of the overall likelihood of core damage and radioactive material releases; and, if necessary, reduce the overall likelihood of core damage and radioactive releases by modifying hardware and procedures that would help prevent or mitigate severe accidents.

The specific objectives are to:

- Identify the potential accident sequences that contribute to the overall core damage frequency.
- Identify cost-effective modifications to the plant design, operating procedures, training or maintenance procedures that would reduce the likelihood of any accident outliers that are identified.
- Maximize participation in the evaluation process by NSP personnel.

- Provide a well organized and clearly written summary of the Monticello IPEEE to facilitate communication of the results to both the NRC and NSP, as well as to serve as a tool for communicating the results to interested members of the public.

- Develop the risk-based tools and methods and the associated documentation to support resolution of future regulatory, safety, or operational issues for Monticello.

1.2 Conformance with Generic Letter and Supporting Materials

NSP's probabilistic risk assessment (PRA) group has been actively involved with the IPEEE process since its inception. Lead responsibility for the IPEEE effort was assigned to the PRA group, which is a part of NSP's Licensing and Management Issues department. The PRA group directed all aspects of the analysis, with general consulting services provided by TENERA, Inc., and EQE International. The PRA group directed the effort, coordinated the work with affected members of the plant staff, was involved in various aspects of the analysis process, and has actively worked with the consultants to ensure the transfer of technology to NSP, so that future applications of the IPEEE can be performed by NSP personnel with the need for only limited external resources. Further details of the organization are provided in section 3 of this report.

This report documents NSP's completion of the IPEEE in accordance with Supplements 4 and 5 to Generic Letter 88-20. A comprehensive review of the IPEEE work was performed by NSP personnel. A review team composed of plant staff and corporate personnel of various disciplines reviewed this report prior to its publication as described in section 2.

In addition to the reviews of the completed analyses, various reviews and validations were performed as part of the analytical process. Walkdowns supporting each of the topics reviewed in the IPEEE were performed to confirm input assumptions and final conclusions. References to these walkdowns are provided in each appendix.

For the seismic IPEEE, screening of the capacity of SSCs was performed at 0.3g in accordance with EPRI NP-6041-SL. The critical safety functions necessary during an accident were reviewed to identify the key systems used to accomplish those functions. Through work performed as a part of the SQUG program, it was demonstrated that there is high confidence that multiple systems would be available to accomplish core cooling and containment pressure control for seismic events as large as the SSE. It was further shown that even if a conservative assumption were made that all SSCs not screened out at 0.3g were unavailable, sufficient equipment would remain available to accomplish core cooling and containment functions.

Sensitivity analyses were performed for the internal fire analysis to identify important fire areas, operator actions, and plant components that drive the potential risk associated with internal fires. The results of these sensitivity studies are presented in the discussion of accident sequence results in Appendix B.

The majority of the assessment of other external events (Appendix C) did not require uncertainty or sensitivity analyses, as most issues could be resolved by comparison with the NRC's standard review plan. When additional probabilistic or deterministic analyses were needed for these other external events, bounding analyses or sensitivity studies were performed to address specific uncertainties.

1.3 Structure of IPEEE Report

NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," identifies reporting guidelines for IPEEE submittals. A cross-reference between the headings in the standard table of contents suggested in NUREG-1407 and this report is provided in Table 1. The most notable difference between the suggested format and that used in this report is that the individual evaluations for severe accident vulnerabilities for seismic events, internal fires, and other external events are contained in separate appendices at the end of this report. The seismic margins assessment is in Appendix A; the assessment of internal fires is in Appendix B; and the evaluation of other external events is in Appendix C. Each appendix is designed to stand alone in order to facilitate their separate review.

Table 1

Cross-Reference Between NUREG-1407 Topics and Contents of this Report

NUREG-1407, Table C.1: Topic	Location in this Report
Executive Summary: - Background and Objectives - Plant Familiarization - Overall Methodology - Summary of Major Findings	 - Main report and section 1.1 of each appendix - Section 1.2 of each appendix - Main report and section 1.3 of each appendix - Main report and section 1.4 of each appendix
Examination Description: - Introduction - Conformance with generic letter and supporting material - General Methodology - Information Assembly	 - Section 1.1 of main report - Section 1.2 of main report - Appendix A, section 2 Appendix B and C, section 2.1 - Appendix A, sections 2.1 and 2.2 Appendix B, section 2.3 Appendix C, sections 2.2 through 2.4
Seismic Analysis (seismic margins assessment)	Appendix A
Internal Fires Analysis	Appendix B
High Winds, Floods and Others	Appendix C
Licensee Participation and Internal Review Team	Section 2 of main report
Plant Improvements and Unique Safety Features	Appendix A, section 2.0 (and subsections) Appendix B, sections 2.11 and 2.15 Appendix C, sections 2.2 through 2.4
Summary and Conclusions	Section 3 of Main Report (Overall) and Appendix A, section 2.5 Appendix B, section 2.15 Appendix C, section 3

2. UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

2.1 IPEEE Program Organization

The director of NSP's Licensing and Management Issues department has the overall review and approval responsibility. The members of NSP's probabilistic risk assessment (PRA) team report to the director of Licensing and Management Issues and, as a team, act as the PRA/IPEEE program manager. The PRA team is responsible for the details and overall project management for all PRA and IPEEE analyses at NSP. Four PRA team engineers worked on the Monticello PRA/IPEEE, all of them located at the plant site. The experience of these team members includes the following:

- Three of the four engineers have obtained SRO licenses or are certified.
- Two have participated in the development and application of the Monticello and Prairie Island PRAs since inception of the IPE program in 1987.
- All are active participants in various industry groups related to resolving severe accident issues, including NUMARC's Severe Accident Working Group, the BWR Owners Group, its Integrated Risk Based Regulation subcommittee, and the MAAP Users Group.
- Collectively, these team members have over 65 years of nuclear power plant experience.

TENERA, Inc. and EQE International, provided consulting services in support of the IPEEE program. TENERA has worked with the NSP PRA group since the inception of NSP's IPE program. EQE International provided expertise in the seismic/ structural engineering area having extensive industry experience in seismic PRAs, seismic margin assessments, and USI A-46 programs. TENERA and EQE actively worked with the PRA team specifically to ensure the transfer of technology to NSP's PRA group at Monticello.

2.2 Composition of the Internal Review Team

In addition to the involvement by NSP's PRA team in the IPEEE program, various plant organizations were involved throughout the evaluations as well as during an internal review of the IPEEE results. For the IPEEE, a review team was selected that would provide a thorough and diverse consideration of both the assumptions in the analyses and the results and conclusions produced by those analyses. This review took advantage of specific organizations that have related programs underway to review the IPEEE results. Examples include the Monticello Design Basis Standards group, the Seismic Qualification Utilities Group (SQUG) effort associated with Unresolved Safety Issue (USI)A-46, and the fire protection group associated with maintaining the Appendix R analyses.

3. IPEEE INSIGHTS AND RECOMMENDATIONS

3.1 Conclusions and Insights from the IPEEE analyses

The external events examination was conducted in three distinct phases: seismic, internal fires, and other external events. Each of these individual studies is described in the appendices of this report. The following summarizes the conclusions of these assessments, including the specific insights and recommendations for plant improvements.

3.1.1 Seismic Analysis

NSP had originally planned to respond to Generic Letter 88-20, Supplement 4, by conducting a seismic PRA. Much of this effort was accomplished (i.e., walkdowns and initial screening) when the NRC issued Supplement 5 to the Generic Letter. NSP elected to change its approach in accordance with Supplement 5 and has completed the analysis of seismic events in the form of a reduced scope seismic margins assessment with a focus on a few known weaker, but critical, components. The majority of the components included in the assessment were determined to meet the screening criteria established in EPRI NP-6041-SL (A Methodology for Assessment of Nuclear Power Plant Seismic Margin). This result in itself indicates that most components have a relatively high seismic capacity. The remaining components; i.e., those not meeting the screening criteria, were evaluated further and determined either to be adequate for the safe shutdown earthquake (SSE), or are to be upgraded under the SQUG program which is being conducted in parallel with the IPEEE effort. Overall, it was concluded that there is no significant plant vulnerability to severe accidents attributable to seismic events at Monticello.

It is important to point out that the seismic analysis conducted as part of the IPEEE program was done in conjunction with the efforts at Monticello to address seismic issues associated with the USI A-46 program. This coordination of programs is the basis for crediting certain components that will be upgraded to the SSE level under the SQUG program. An additional significant result is that the IPEEE analysis showed that, even assuming the failure of those components that were not screened out at the 0.3g level, these seismically induced failures would not be expected to lead to an inability to cool the core. In each case, additional random failures of equipment are necessary before inadequate core cooling would be expected, and substantial time is available (on the order of days) to restore containment heat removal before core cooling would be threatened.

Other significant conclusions of the seismic analysis include:

- The seismic walkdowns performed as part of the IPEEE found most of the components and structures reviewed to be seismically adequate (i.e., suitably anchored and/or seismically rugged). Those items that could be considered potentially vulnerable were subjected to the more rigorous seismic analyses referred to above;

- Concrete block walls were either screened from further consideration because their failure would cause no adverse consequences, or they were further evaluated and found to have sufficient seismic capacity;

- The review of relays revealed that there were potential "bad actor" relays that were included in the scope of the IPEEE. However, the configuration and/or consequences of chattering of these relays is such that only relays already included in the scope of the SQUG program are of any significance;

- Few flat bottomed tanks fell under the scope of the seismic IPEEE. Those that did were either screened or shown to have limited consequences were they to fail;

- A review of containment response reveals no conditions that are unique to seismic events or that have not already been evaluated as part of the internal events PRA (IPE).

3.1.2 Internal Fires Analysis

The total plant risk due to fires at the Monticello nuclear generating station was calculated to be less than 7.8E-6 core damage events per year. Eighty-three percent of the plant risk associated with internal fires can be traced to seven rooms or burn areas: (1) the main control room; (2) turbine building 931' (fire zones XII/17, 19A and 19B); (3) the MCC 133/feedwater pump area; (4) the cable spreading room; (5) the reactor building 935/962' west; (6) the lower 4KV switchgear room and; (7) the Division II area of the EFT building. These results are consistent with the results of recent fire PRAs performed at other plants, in terms of both absolute value and percentage contribution to total plant risk.

The fire IPEEE accident sequence quantification includes a number of conservatisms. For example, fires were always assumed to completely engulf the area in which they started. Automatic or manual fire suppression was not credited except in the MCC/feedwater pump area, the main control room and the cable spreading room. Further, repair activities were only applied to accident sequences in which a very long time was available to effect repairs, and then only to those components not damaged by the fire. When repair actions were credited, the recovery of only a single failed component was assumed even if there were multiple failures to which recovery could be applied. Systems were also assumed to fail in certain areas to limit the effort required to perform cable tracking. Therefore, the methodology, while yielding useful reliable results, gives core damage frequencies that are considered to be upper bounds.

The relatively low plant risk due to fires results in large part from Monticello's plant-specific implementation of the requirements of 10CFR50, Appendix R. These requirements include separation of alternate or redundant trains of safe shutdown equipment, installation of fire barriers, and installation of an alternate shutdown system located outside the main control and cable spreading rooms. The total risk due to fires is also kept low by the administrative control of transient combustibles.

Core damage will not occur unless random equipment failures unrelated to damage caused by the fire also occur. This fact, in conjunction with the low overall core damage frequency due to fires, precluded identification of any risk-significant insights. However, one potential improvement to this study was noted. This improvement addresses conservatisms in the current analysis. Because of the location specific nature of the analyses and the effort involved in performing these analyses for all areas throughout the plant, manual suppression effects were analyzed and credited only in the control room. If future applications warrant, the effects of manual suppression can be analyzed and credited on a location-by-location basis. Similarly, CRD and the main condenser were only credited for fires in a few areas. These systems may be credited on a location-by-location basis by verifying that cables/equipment required by these systems are not located in the locations in question. Incorporation of this improvement may result in a significant reduction in overall calculated plant risk. These calculational conservatisms should be considered before considering any plant modification.

3.1.3 High Winds, Floods, and Others

The assessment of other external events shows that there is no external event (other than internal fires and seismic events) that may be a safety concern to the Monticello plant. No vulnerabilities were identified and the screening criteria contained in NUREG-1407 and Generic Letter 88-20, Supplement 4, were satisfied for all events. A simple walkdown confirmed these results.

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Appendix A
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Seismic Analysis

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A.1. INTRODUCTION

A.1.1 Background

This report documents Northern States Power Company's (NSP's) response to Supplement 4 of Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," for the Monticello Nuclear Generating Plant. The assessment described in this appendix addresses seismic events. The analysis and its results, which are described in the following sections, provide insights with respect to the response of the Monticello plant to a seismic event. As described in Supplement 5 to Generic Letter 88-20, an evaluation equivalent to a reduced scope seismic margins assessment was performed for Monticello with an additional focus on a few key critical components.

A.1.2 Plant Familiarization

The Monticello Nuclear Generating Plant is a low power-density boiling water reactor. General Electric Company designed the plant and supplied the nuclear steam supply system, the turbine generator unit, and their related support systems. Bechtel Corporation constructed the plant. The design is identified by General Electric as a "BWR-3" with a Mark I containment. The reactor core produces 1670 MWt with an electrical output of 545 MWe, using 484 fuel assemblies. The plant is located in Monticello, Minnesota. Construction started on June 19, 1967, and full commercial operation began on June 30, 1971.

The original design considered seismic events in the design of Class I systems, structures and components. Chapter 12.2.1 of the Monticello USAR defines Class I as "structures and equipment whose failure could cause uncontrolled release of excessive amounts of radioactivity or which are vital to safe shutdown of the plant and the removal of decay heat." Some Class II structures house Class I equipment. Although not subjected to a dynamic analysis, such Class II structures were evaluated for Class I loads in the areas housing Class I loads. Class I structures and equipment are designed for a horizontal ground acceleration of 0.06g for the Operating Basis Earthquake (OBE) and 0.12g for the Safe Shutdown Earthquake (SSE). Seismic evaluations of electrical equipment anchorage and masonry walls were performed in the early 1980s. These evaluations resulted in modifications that increased the seismic capacity of many electrical components and certain masonry walls.

A.1.3 Overall Methodology

NSP originally planned to respond to Generic Letter 88-20, Supplement 4 [1], by performing a seismic probabilistic risk assessment (PRA) for Monticello. The walkdowns and screening evaluations of essential structures and equipment were performed following procedures applicable to a focused scope plant, which was how Monticello was categorized in NUREG-1407. In accordance with Supplement 5 to Generic Letter 88-20 [2], NSP subsequently elected to complete the Monticello seismic IPEEE by conducting the equivalent of a reduced scope seismic margins assessment with an additional focus on a few key components.

The overall methodology for the Monticello seismic IPEEE thus consists of the following steps:

- Systems and components considered in the seismic IPEEE were identified based on insights from the internal events PRA.
- A walkdown of key plant structures and systems was performed following the EPRI NP-6041-SL [3] procedures for a focused scope seismic margin evaluation to screen seismically rugged structures and components from further review.
- Evaluations were performed following the procedures for a focused scope seismic margin assessment to further screen structures and components from further review.
- A list was compiled of components that did not meet the screening criteria in the preceding two steps. This compilation of unscreened components represents a potentially conservative set of outliers for a reduced scope seismic margin assessment.
- The unscreened components were dispositioned following the requirements for a reduced scope seismic margin assessment by either (1) the results of the Monticello Seismic Qualification Utilities Group (SQUG) program (part of the Unresolved Safety Issue (USI) A-46 [4] effort), or (2) a review of the effect of component failure on plant systems.
- An evaluation of the severe accidents which may be initiated by an earthquake was done to ensure that the final list of equipment which would be available following a safe-shutdown earthquake (SSE) is in fact sufficient to provide multiple means of bringing the plant to safe shutdown. A further evaluation was done to show that the plant could also be brought to safe shutdown using only the equipment which was found to be seismically rugged during the walkdown and screening evaluations.

These steps are described in more detail in Section A.2.

A.1.4 Summary of Major Findings

No one safety function dominates the results of the Monticello seismic margins assessment. The safety functions considered for the IPEEE are similar to those used to define the accident sequence types quantified in the IPE:

- Reactivity control
- High pressure injection
- Reactor depressurization
- Low pressure injection
- Containment pressure control
- Important support systems

Most components included in the seismic margins assessment for Monticello which support these functions have relatively high seismic capacities. The components that do not meet the more conservative focused scope seismic margins assessment screening criteria and contribute to the safety functions noted above are summarized in Table A.1.

Each of the components identified in Table A.1 were shown to be adequate at the SSE or are to be addressed under the SQUG program. Even assuming the failure of those components that are not to be upgraded, these seismically induced failures would not be expected to lead to an inability to cool the core. In each case, additional random failures of equipment are necessary before inadequate core cooling would be expected and substantial time is available (days) to restore containment heat removal before core cooling would be threatened. A review of containment response reveals no conditions that are unique to seismic events or that have not been evaluated as a part of the internal events IPE. It is therefore concluded that the Monticello plant has no vulnerability to seismically induced severe accidents.

Table A.1 Monticello Seismic IPEEE: Summary of Major Findings

Function/Components	Failure Mode	Postulated Effect of Failure	Conclusions
Reactivity Control			All SSCs* screened
High Pressure Injection	Impact with panel C-289A	Contains relays for high reactor level trip of HPCI and RCIC	Acceptable at the SSE.
Relay Panel C30			
Low Pressure Injection			
Relay Panel C32	Impact with HVAC duct	Contains relays for LPCI injection valves MO-2014 and MO-2015.	To be upgraded under SQUG program.
RHR Pumps B & D, and Core Spray Pump B	Impact with RHR Room B HVAC unit.	Could affect operation of one division of low pressure injection.	Acceptable at the SSE.
RHR and Core Spray Pumps	Grouted in place anchor bolts	Could affect both divisions of LPCI and Core Spray.	Acceptable at the SSE.
Reactor Depressurization			All SSCs Screened
Containment Pressure Control			
RHR Pumps B & D	(See Low Pressure Injection)		
RHR Pumps	(See Low Pressure Injection)		
Support Systems			
AC Power			
DG11 & 12 Air Receivers	Low pretension of U-bolts	Could result in EDG failure to start	To be upgraded under SQUG program
4KV Bus 16	Buckling of top brace	Could result in loss of Division II AC	Acceptable at the SSE
MCC 43A	Anchorage	Could result in loss of DG12 support systems	Acceptable at the SSE
DC Power			
Battery Chargers D70, D80, D90	Weak axis bending of channel supports	Loss of Div II 250VDC after 4 hrs (HPCI and partial loss of DC to SRVs E, F, G & H).	Acceptable at the SSE
Battery Chargers D52, D53, D54	Weak axis bending of channel supports	Loss of Div I 250 VDC after 4 hrs (RCIC and partial loss of DC to SRVs E, F, G & H).	Acceptable at the SSE
MCC D312	Anchorage	Power Supply for HPCI	Acceptable at the SSE
MCC D311	Anchorage	Power Supply for RCIC	Acceptable at the SSE
Nitrogen			All SSCs for bottled N_2 screened
EDG Service Water			All SSCs screened
Emergency Service Water			All SSCs screened
RHR Service Water			All SSCs screened
Miscellaneous			
Control Room Ceiling	Lighting not safety-wired	Inhibit operator actions such as: - Emergency depressurization - Low pressure injection initiation	To be corrected under SQUG program

* SSC = System, structure, or component

A.2. SEISMIC ANALYSIS

A seismic margins assessment of the Monticello Nuclear Generating Plant was conducted between 1992 and 1995 to address the requirements of Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," dated June 1991. In accordance with Supplement 5 to Generic Letter 88-20, the assessment is equivalent to a reduced scope seismic margins assessment with an additional focus on a few key components. These components include relays, block walls, flat-bottom tanks, and other components identified during the plant walkdowns.

The Monticello seismic margins assessment follows the guidance of EPRI NP-6041-SL with additional input from the internal events probabilistic risk assessment (PRA). This assessment included the following elements:

- System, structure and component success path selection
- Plant walkdowns
- Structure seismic response analysis
- Component screening
- Seismic margin assessment

The success paths for the Monticello seismic IPEEE were derived from the logic models developed for the internal events PRA. Active components of all systems which could be available following a loss of offsite power were included on the equipment list used during the walkdown and screening activities. With this approach, multiple potential success paths were identified for each safety function. The list was further supplemented with passive components which were not modeled in the internal events PRA, such as tanks, heat exchanger, panels, cabinets, and support structures.

The plant walkdowns were conducted following the guidelines for seismic margin assessment presented in EPRI NP-6041-SL. The walkdown was performed to screen seismically rugged structures and components from further review, to identify the potential failure modes and system interactions for components that could not be screened from further review during the walkdown, and to obtain data for use in subsequent evaluations. The walkdown teams included systems analysts and seismic capability engineers. The walkdown procedures followed EPRI NP-6041-SL guidelines. The walkdown screening was based on a seismic margin earthquake having a peak 5% damped spectral acceleration of 0.8g or less. This screening level is applicable to a focused scope plant and is conservative for a reduced scope plant. The results of the walkdowns were recorded on data sheets for future reference. Seismic issues requiring further review were identified.

Seismic analyses of the reactor, turbine, and control buildings were performed to obtain structure seismic loads and the in-structure response spectra for subsequent evaluations of structures and equipment. Earthquake acceleration time-histories were generated with response spectra matching specified target ground spectra. The site response was analyzed to obtain strain-compatible free-field soil properties. Foundation impedances and scattering functions were calculated using current methods. Existing structure models were used with minor modifications. Soil-structure interaction analyses were performed to obtain seismic responses for the selected buildings.

Conservative evaluations were performed to further screen structures and components from more detailed review. These screening evaluations were performed for a seismic margin earthquake having a peak 5% damped spectral acceleration of 0.8g or less, following the guidelines of EPRI NP-6041-SL. This screening level is conservative for a reduced scope margins assessment.

Based on the walkdown and the conservative screening evaluations, nearly all of the essential components were screened from further review. Some of the components were conservatively kept on the list of items for further review based on the seismic margin assessment guidelines for a focused scope plant. These components were further reviewed following the NUREG-1407 guidance for seismic margin assessment of reduced scope plants. Many of these components were found to be acceptable for the SSE by the Monticello SQUG program, or will be seismically upgraded for the SSE by the SQUG program.

Those components which were not eliminated from further consideration during the walkdown and screening evaluations were dispositioned by a systems analysis that considered the effect of failure of these components, as well as determining whether other systems would be available to perform the critical safety functions needed during an accident following a seismic event.

A.2.1 Plant Systems

The plant systems considered in the seismic IPEEE are a subset of those considered in the IPE. An earthquake could reasonably produce a loss of offsite power (LOOP) or a small-break loss of coolant accident (SLOCA) initiating event. The seismic portion of the IPEEE focused on the frontline and support systems that would be called upon to prevent core damage for these two initiating events. The systems considered are listed below by the functions they support.

- Reactivity Control
  - Reactor protection system (RPS)
  - Alternate rod insertion (ARI)
  - Standby liquid control (SLC)
- Reactor Pressure Control
  - Reactor pressure vessel (RPV) safety/relief valves (SRVs)
- High Pressure Coolant Makeup
  - High pressure coolant injection (HPCI)
  - Reactor core isolation cooling (RCIC)
- RPV Depressurization
  - SRVs
- Low Pressure Coolant Makeup
  - Low pressure coolant injection (LPCI)
  - Core spray (CS)
  - RHR service water (RHRSW)
  - Fire protection system (FPS)

- Containment Pressure Control
    - Residual heat removal (RHR)
    - RHR service water (RHRSW)
    - Fire protection system (FPS)

This section discusses the plant systems considered in the seismic IPEEE and the specific equipment comprising those systems.

A.2.1.1 Plant Frontline Systems Included in the IPE

A discussion of the frontline plant systems included in the seismic IPEEE by functional area is included in this section. A brief discussion of those systems considered in the IPE but not credited in the IPEEE is also provided.

A.2.1.1.1 Reactivity Control

Reactor Protection System (RPS)

Reactor protection system (RPS) instruments monitor key plant parameters to determine whether the plant processes are within the bounds of normal operation.

A seismic event sufficiently large to cause equipment damage is expected to result in an RPS trip signal from a variety of causes. Potential conditions in the plant likely to cause a reactor trip following a seismic event include loss of off-site power or small loss of coolant accidents and their resultant plant conditions.

Supplement 5 to Generic Letter 88-20 excludes evaluation of reactor internals from consideration. In the Monticello IPEEE, consideration was given to other equipment necessary to assure a seismic event would not cause a failure to scram.

Alternate Rod Insertion (ARI) System

Alternate rod insertion (ARI) is a means of control rod insertion that uses the hydraulic control units and control rod drives as they normally function, but which is triggered by separate and diverse logic from the RPS. Its purpose is to provide a redundant mechanism for reactor scram in the unlikely event that electrical failure of the RPS or its sensors does not result in rod insertion. The ARI initiation signals of high vessel pressure or low water level are used to open separate solenoid valves that cause the scram pilot air header to depressurize. Depressurization through the ARI valves takes approximately 25 to 30 seconds, after which the reactor is shut down by rod insertion. ARI is assumed not to be effective if the failure to scram is due to a mechanical problem which prevents control rod insertion. Successful actuation of ARI requires that one of two solenoids energize to vent the air supply to the scram valves.

Standby Liquid Control (SLC) System

The SLC system can independently shut down the reactor by injecting sodium pentaborate solution. Successful SLC operation requires that one of the two positive displacement pumps be started and that either of the two squib valve explosive charges be actuated to provide a path to the reactor.

A.2.1.1.2 Reactor Pressure Control

Reactor Vessel Pressure Relief System

The reactor vessel pressure relief system consists of eight relief valves, all located on the main steam lines within the drywell. The relief valves are self-actuating at 1109 psig. Analyses for the Monticello internal events IPE showed that one SRV is sufficient to provide reactor overpressure protection. Consideration is given to the potential for an SRV failing to close, which would depressurize the reactor and cause the loss of turbine-driven high pressure injection systems.

A.2.1.1.3 High Pressure Injection

High Pressure Coolant Injection (HPCI) System

The HPCI system consists of a steam turbine assembly that drives a constant-flow pump and includes related piping, valves, and instrumentation. The system automatically initiates upon sensing high drywell pressure or low-low reactor water level. The HPCI pump can draw suction from either the suppression pool or the condensate storage tanks (CSTs), with interlocks to ensure that only one source is aligned at any given time. The HPCI system can provide primary coolant makeup at a rate of approximately 3000 gpm, which is sufficient to maintain level above the top of the fuel for transients in which makeup is required due to decay heat, small LOCAs, and ATWS events.

HPCI operation depends on the Division II DC batteries. With the exception of the battery chargers for these batteries, HPCI is independent of the availability of AC power sources.

While the primary source of HPCI suction is the CSTs, the CSTs were assumed to be lost as a result of the seismic event. This leaves the suppression pool as the suction source for HPCI.

Reactor Core Isolation Cooling (RCIC) System

The RCIC system consists of a steam turbine assembly that drives a constant-flow pump and includes related piping, valves and instrumentation. The system automatically starts upon sensing low-low reactor water level using a one-out-of-two-taken-twice logic. The design flow of the RCIC system is approximately 400 gpm. Since this is relatively low flow, RCIC is considered to be adequate for transients requiring makeup due to decay heat generation, but not for ATWS or LOCA events.

RCIC operation depends on Division I DC batteries. With the exception of the battery chargers for these batteries, RCIC is independent of the availability of AC power sources.

As with HPCI, the primary source of water for RCIC in this analysis is the suppression pool, since the CSTs are assumed to fail following a seismic event.

Control Rod Drive (CRD) System

No credit was taken in this analysis for the use of the CRD system to inject makeup flow to the reactor. Normal flow through the CRD system for makeup to the reactor would be from the condensate system; however, since it is assumed that the earthquake would cause a loss of offsite power, the condensate pumps would not be running. The backup source for the CRD system is the CSTs; since these tanks are assumed to fail during the earthquake, no suction source is available. Moreover, the CRD system is automatically load shed from the emergency diesel generators on loss of off-site power with a concurrent ECCS initiation signal. No credit was taken for operator action to bypass this load shed signal to permit operation of the CRD pumps, even though this guidance is provided in the emergency operating procedures.

Feedwater (FW) System

The feedwater system is not available following a seismic event, which is assumed to cause a loss of off-site AC power. Feedwater is therefore not credited in the seismic IPEEE.

A.2.1.1.4 Reactor Depressurization

Reactor Vessel Pressure Relief System

The reactor vessel pressure relief system consists of eight safety/relief valves (SRVs), all located on the main steam lines within the drywell. All of these valves discharge into the suppression pool; there is no discharge directly into the drywell. All eight are available to depressurize the reactor. Three are associated with the automatic depressurization system (ADS). All SRVs can also be operated individually with remote manual controls in the main control room. A single SRV is sufficient to reduce reactor pressure below the shutoff head of the low pressure pumps if the reactor has tripped.

Short-term operation of the SRVs is independent of AC power because all eight SRVs have DC solenoid valves and accumulators to provide pneumatic pressure for opening. Long term operation of all eight SRVs relies on AC power for battery charger operation.

It is assumed in the Monticello seismic IPEEE that any time the automatic depressurization system timer is actuated, ADS is manually inhibited as directed by the emergency operating procedures. Reactor depressurization is therefore considered to require manual initiation to be successful. Operator action is assumed to be required within 30 minutes of the loss of high pressure injection systems.

A.2.1.1.5 Low Pressure Injection

Low Pressure Coolant Injection

Low pressure coolant injection (LPCI) is an operating mode of the residual heat removal (RHR) system. The LPCI mode of the RHR system is designed to inject water into the reactor vessel to restore and maintain water level in the event of a LOCA. LPCI also can provide coolant inventory makeup after the reactor vessel has been depressurized to less than 325 psig through the SRVs.

In the LPCI mode of operation, the RHR pumps take suction from the suppression pool and discharge the water into the reactor vessel via the reactor recirculation loop piping. Each pump has a dedicated suction line from the suppression pool and each loop has a dedicated injection path to the reactor vessel through one of the reactor recirculation loops. The Monticello RHR system has "loop selection logic" which is designed to automatically select one recirculation loop for LPCI.

In the LPCI mode the RHR pumps are designed to start automatically upon receipt of any one of the following signals: (1) a low-low reactor vessel water level signal with low reactor pressure, or (2) a high drywell pressure signal, or (3) low-low reactor level sustained for 20 minutes. Any one of the four pumps is sufficient to provide reactor makeup. The suction of the RHR pumps is normally aligned to the suppression pool.

Core Spray (CS) System

The core spray system is designed to automatically spray water onto the top of the core at a sufficient rate to cool the core and limit fuel clad temperatures in the event of a LOCA. In addition, the core spray system can provide coolant inventory makeup if the reactor vessel is depressurized to less than 350 psig through the SRVs.

The core spray system consists of two independent loops, each containing a core spray pump, a sparger ring with spray nozzles, and the necessary piping, valves and instrumentation. The signals which automatically start the LPCI system also initiate core spray. A single train of core spray is sufficient to provide reactor makeup even under LOCA conditions. The suction of the core spray pumps is normally aligned to the suppression pool.

RHR Service Water/Fire Protection Cross-Tie to LPCI

The RHR service water (RHRSW) and fire protection systems can be cross-tied to the RHR system as an emergency low-pressure injection source. Manual valves in the RHRSW-to-LPCI cross-tie path must be opened to provide a flowpath from the RHRSW or fire pumps to the discharge side of RHR heat exchanger A. Use of this alignment requires that the reactor vessel remain depressurized below the shutoff head of these low pressure systems (125 psig). This mode of reactor makeup would be necessary in the seismic IPEEE only if all other means of high pressure injection and low pressure injection were unavailable.

A.2.1.1.6 Containment Pressure Control

Residual Heat Removal (RHR) System

The RHR system is divided into two loops. Each loop contains two pumps and one heat exchanger with an associated heat exchanger bypass valve. Each RHR pump has a rated capacity of approximately 4000 gpm. The RHR system operates in the following modes to remove decay heat as described below:

RHR Suppression Pool Cooling Mode

In the suppression pool cooling mode of RHR system operation, the RHR pumps take suction from the suppression pool and discharge the heated suppression pool water through the shell side of the RHR heat exchanger and back into the suppression pool. Flow from the RHR service water system passes through the RHR heat exchanger tubes and cools the suppression pool water.

The suppression pool cooling mode of RHR is the preferred method of decay heat removal if the main condenser is unavailable. Suppression pool cooling may be initiated at any time following reactor scram, since it is independent of the reactor vessel pressure. In the suppression pool cooling mode, one loop of the RHR system (one RHR pump and the associated RHRSW pump train) is adequate for heat removal from the suppression pool for all initiating events.

RHR Shutdown Cooling Mode

In the shutdown cooling mode of operation, the RHR pumps take suction directly from the reactor vessel and discharge the heated reactor coolant through the shell side of the RHR heat exchanger and back into the reactor vessel. Flow from the RHR service water system passes through the RHR heat exchanger tubes and cools the reactor coolant. Because of the pressure rating of the suction piping, the shutdown cooling mode is not placed into operation until the reactor vessel pressure is less than 40 psig. Shutdown cooling was considered in the seismic IPEEE as an alternate means of providing decay heat removal redundant to suppression pool cooling.

Drywell and Wetwell Spray

The two other modes of RHR -- drywell sprays and wetwell sprays -- can also provide containment pressure control. Because these modes rely on much of the same equipment that is needed for the shutdown cooling and suppression pool cooling modes of RHR, it is assumed that if these RHR cooling modes are not available, then RHR is also not available in the spray modes. However, even if RHR is unavailable, the drywell and wetwell spray headers can be supplied from either RHRSW or the fire protection system. This requires that RHRSW or the fire pumps be aligned to the discharge side of the RHR A heat exchanger. This form of containment pressure control would be initiated only if all modes of RHR were unavailable.

Main Condenser

The main condenser, including supporting balance-of-plant systems, is assumed to be unavailable following a seismic event due to its reliance on off-site AC power.

Containment Vents

The wetwell and drywell vent and purge valves require instrument air to open. Instrument air was not credited following a seismic event, as it is a non-seismic system that has components located throughout the plant that, for simplicity, were not included in the walkdown. A hard-piped vent was installed in response to Generic Letter 89-16. The hard-piped vent is operated with nitrogen and includes a solenoid valve that is dependent on Division II 250 VDC. The hard-piped vent would be used for containment heat removal only after all modes of RHR had failed. It would be used to keep the containment below its design pressure of 56 psig, which would not be reached until more than a day after the earthquake, even assuming no other means of heat removal was available.

A.2.1.2 Support Systems Included in the IPE

AC Electrical Distribution System

Off-Site AC Power

The seismic events considered for the IPEEE were those sufficiently large to cause a loss of off-site power. If the seismic event did not cause a loss of off-site power, sufficient systems were assumed to remain available such that the potential for core damage could be considered to be covered by the internal events PRA.

On-Site AC Power

The on-site AC power system is made up of two emergency diesel generators and the plant AC distribution system. The AC distribution system is made up of six 4kV buses feeding the large motors, and various 480V load centers. Loss of voltage or degraded voltage on the essential buses will start the emergency diesel generators and initiate load shedding to allow the diesels to supply their respective buses.

A third diesel generator can also be cross-tied to supply the battery chargers by aligning it to the chargers through a series of manual cross-ties. This analysis assumes that the operator does not accomplish this manual alignment within the first few hours following a seismic event. The third diesel was not evaluated as a part of the seismic IPEEE.

DC Electrical Distribution System

For the purposes of this analysis, the DC power system consists of two divisions of 250VDC batteries and two divisions of 125VDC batteries. The major loads powered by the 250VDC batteries considered in this analysis were HPCI, RCIC, and the uninterruptible power supplies for instrumentation and system controls. Similarly, the major loads powered by 125VDC batteries were the SRVs, HPCI and RCIC control power, 4KV motor control power, power for starting and loading the emergency diesel generators, and miscellaneous control power. As in the internal events PRA, battery capacity during blackout is assumed to be four hours without load shedding.

Emergency Service Water

The purpose of the emergency service water (ESW) system is to supply cooling water to the emergency filtration train (EFT) system, ECCS pump motor coolers, and ECCS room coolers. The pumps start automatically. As in the internal events PRA, the RHR motor coolers were the only components specifically assumed to be dependent on the ESW system. There are two ESW pumps. Each ESW pump supplies water to specific RHR pumps. Even without ESW, the RHR pumps could operate steadily for several hours or could be operated intermittently and staggered to prolong pump operation.

Service water is a backup to the ESW system, but it is load shed if there is a loss of offsite power coincident with an ECCS initiation signal. Since the seismic event is assumed to cause a loss of offsite power, no credit was taken for service water backing up ESW.

EDG Emergency Service Water (EDG ESW)

The purpose of the EDG ESW system is to supply cooling water to the diesels. There is one pump for each diesel, and this pump is required for the diesel to operate. Service water is a backup to the EDG ESW system, but it is load shed if there is a loss of off-site power coincident with an ECCS signal. Because the seismic event is assumed to cause a loss of offsite power, no credit is taken for this backup capability. The EDG ESW pump is started automatically when the diesel reaches 125 RPM and the bus is energized.

Residual Heat Removal Service Water (RHRSW) System

The RHRSW system is the heat sink for various modes of RHR system operation. One RHRSW pump is required for successful system operation. The system consists of two separate loops with two pumps each. The RHRSW system can also be used as a low pressure injection system or a source of water for drywell and wetwell sprays by manually aligning the discharge of the pumps to Loop B of RHR downstream of the RHR heat exchanger.

A.2.1.3 Supporting Components Included in the IPEEE

The seismic IPEEE included distributed systems whose failure could cause a loss of function of the systems listed above, such as piping, cable trays, and conduit. The HVAC ducting need not function to ensure performance of essential systems, but the potential for the HVAC ducting to interaction with other systems during an earthquake was included in the seismic IPEEE.

A.2.2 Plant Walkdown

The objectives of the plant seismic walkdowns were (1) to identify any equipment having sufficiently high seismic capacity that no further review was needed, (2) to identify the potential failure modes and system interactions for equipment that could not be screened from further review, and (3) to obtain data on equipment and structures for use in detailed evaluations of the potential failure modes and system interactions. Preparation for the initial plant walkdown is described in Section A.2.2.1. General descriptions of the initial and final plant walkdowns, including procedures and documentation, are presented in Sections A.2.2.2 and A.2.2.3. Significant walkdown findings are identified in Section A.2.2.4.

A.2.2.1 Pre-Walkdown Preparation

Activities performed prior to the initial plant walkdown included (1) information collection and review, (2) equipment list review, (3) identification of equipment locations, and (4) walkdown data sheet preparation.

Information Collection

Information relevant to the seismic IPEEE was collected. The following categories of plant documentation were obtained prior to and during the plant walkdowns:

- Updated Safety Analysis Report (USAR)
- Structural, architectural, and equipment layout drawings
- Equipment anchorage drawings
- Drawings for selected equipment components
- Specifications for construction of civil structures
- Civil structure material strength test data
- Seismic criteria and analysis reports for building structures
- Geotechnical investigation reports
- I & E Bulletin 80-11 documentation on masonry wall seismic qualification
- Reports on containment analyses for loss-of-coolant-accident and safety-relief valve discharge events
- Seismic evaluations of electrical equipment anchorage, and design calculations and drawings for resulting seismic modifications

This documentation was reviewed prior to the walkdown to obtain an understanding of the plant configuration, design, and construction, vital safety systems, structure response characteristics, and structure and equipment capabilities.

A preliminary listing of essential equipment included in the IPE systems model for the seismic IPEEE was developed by NSP. Additions and deletions of selected equipment components were suggested based upon past experience in seismic margins assessment. Further revisions were identified during the walkdown. The essential equipment that was considered in the seismic margins assessment is listed in Tables A.2.4-1 and A.2.5-1.

Components included in the equipment list were located on the mechanical layout drawings to the extent possible. Prior knowledge of equipment locations helped in planning routes through the plant during the walkdown and identification of components in the field. NSP walkdown team members were highly familiar with equipment locations, thus greatly expediting the walkdown.

Walkdown data sheets were prepared for each of the components on the equipment list. To the extent possible, general information was entered into these data sheets by members of the walkdown team before the walkdown. Such information included the equipment class, description, identification number, location (building, floor elevation, and room number), and manufacturer and model number.

A.2.2.2 Initial Plant Walkdown

The initial plant walkdown was performed by a single team over a duration of eight days in July 1992. The walkdown team consisted of two or three seismic capability engineers and at least one plant engineer. The seismic capability engineers were collectively knowledgeable in the seismic behavior of structures and equipment subjected to strong earthquakes and seismic evaluation of nuclear power plant structures and equipment by probabilistic and deterministic methods. Plant engineers included NSP personnel who were qualified as senior reactor operators (SROs) and familiar with the plant systems.

Essential equipment located in non-radioactive or slightly radioactive areas was inspected in the initial plant walkdown. Structures containing these equipment components were also surveyed, including the reactor building, turbine building, control and cable spreading structure, emergency filtration train building, diesel generator building, and intake structure.

The drywell and its contained components were not surveyed in the initial plant walkdown since the plant was operating. Also, inspection of equipment located in moderately radioactive areas was deferred to a subsequent walkdown.

A.2.2.2.1 Walkdown Procedures

The walkdown followed procedures recommended by EPRI NP-6041-SL. As discussed later in Section A.2.4, screening of structures and components was performed following the recommendations of EPRI NP-6041-SL for a seismic margin earthquake having a 5 percent damped peak spectral acceleration of 0.8g or less. Components were surveyed in the walkdown to ensure that caveats implicit in these screening criteria were satisfied.

Structures

Information necessary for screening or seismic evaluation of civil structures is normally obtained from design drawings rather than walkdowns. A complete set of structural drawings was reviewed to obtain a general understanding of building construction and configuration and to identify any specific data to be obtained during the walkdown. The walkdown was performed to (1) verify that the structures are in general conformance with the design drawings, (2) identify any gross deficiencies that might reduce structure capacities, and (3) confirm that separations between the building structures are as indicated on the design drawings.

Concrete Block Walls

Existing documentation on the Monticello seismic design basis prepared in response to USNRC I & E Bulletin 80-11 was reviewed to obtain an understanding of block wall configuration, construction, and seismic capacity. Data contained in the documentation were used in the walkdowns.

A subset of block walls was targeted for walkdown. These walls were prioritized based on their ratios of seismic demand to seismic capacity obtained by the I & E Bulletin 80-11 analyses. Any walls noted to be specific seismic concerns during the equipment walkdowns were also added to the targeted block wall listing.

Prior to the walkdown, data on block wall constructions, locations, and nearby or attached equipment were obtained from the I & E 80-11 documentation and recorded on the walkdown data sheets. Available block wall construction details included block type and grouting, number of wythes, block and wall thicknesses, and reinforcement.

During the walkdown, block walls lacking failure consequences were screened from further review using the following criteria:

- No essential equipment is supported by or in proximity to the block wall. Some of the block walls are categorized as being safety-related, but the affected equipment is not on the essential equipment list.

- Any essential equipment is located about one wall height away from free-standing walls or one-half wall height away from top-supported walls. It is unlikely that such equipment would be impacted even if the wall were to collapse.

- Even if the wall collapses onto an essential component, loss of component function will not occur. For example, piping or heat exchanger with thick walls and rugged supports may survive wall impact.

For walls that do not meet these criteria, detailed data were recorded on walkdown data sheets for use in subsequent seismic evaluation. Data previously obtained from I & E Bulletin 80-11 documentation were verified to the extent possible. Additional data on spans, boundary conditions, and potential for arching action were also recorded. Photographs supplemented the data sheets.

Equipment and Vessels

The initial plant walkdown surveyed components in the accessible areas identified above. Detailed inspections were performed for numerous components. Other components, while not inspected in detail, were determined to be similar to those for which detailed inspections were performed. Sufficient reviews were performed to establish that similarities, in terms of component construction and anchorage, exist. Any component-specific features, such as anchorage details or systems interaction concerns, were recorded.

Key elements of the component walkdowns included review of component configuration and construction, anchorage, and potential system interactions. These reviews followed the guidelines of EPRI NP-6041-SL.

Configurations and construction details of the components and their supports were reviewed to ensure structural integrity and post-earthquake functionality. Following checklists on the walkdown data sheets (Section A.2.2.2.2), components were inspected to ensure that they possessed adequate seismic design features. These features vary depending on the generic component category, and include attributes such as adequate stiffness and strength of component load paths and supports, and adequate attachment and support for appurtenances. Components were also inspected to identify any seismic vulnerabilities, such as unreinforced cabinet cutouts, unrestrained vibration isolators, and excessive component or attachment weights.

Inspection of component anchorage included verifying that the load paths have adequate stiffness and identification of any specific concerns, such as high shims or excessive concrete cracking in the vicinity of the anchor. Screening of component anchorage strength was deferred until after the initial plant walkdown, at which time quantitative evaluations were performed (see Section A.2.5). Data on component anchorage were recorded for use in these evaluations.

Inspection was performed to identify any systems interaction concerns associated with proximity, Class II-over-Class I interactions, and spray and flooding. Essential components in close proximity to adjacent objects were reviewed for potential damage due to relative seismic motion. Only soft targets, such as gauges or small tubing, were considered to be vulnerable to impact damage. Any Class II-over-Class I interactions associated with a non-essential component falling on an essential component were identified. Potential Class II-over-Class I interactions include failure of concrete block walls or ceiling systems. Any credible sources of spray or flooding that could impair the function of an essential component were identified. Potential sources of spray and flooding include failures of wet fire water piping with threaded joints or mechanical couplings and non-essential tanks. If such sources were identified, further review was performed to identify any mitigating features, such as spray shields or floor drains.

The NUREG-1407 guidance for relay evaluation is to follow SQUG procedures for a plant such as Monticello that is required to address SQUG [8. U.S. Nuclear Regulatory Commission, 1987]. Consequently, no relay walkdown for the Monticello IPEEE was performed, but instead was deferred to the SQUG review. For a reduced scope seismic margins assessment, this is sufficient to meet the NUREG-1407 guidance.

Although the detailed relay review was deferred for completion under the SQUG program, it was decided as part of the seismic IPEEE work to pursue additional information concerning the consequences of relay failure to assist Monticello's SQUG program. The objective of this part of the evaluation was to obtain information concerning the functional impacts of selected relays, and to generate a relative ranking of important relays as measured by their impact on core damage frequency. This involved creating a list of relays, including their manufacturer and model type where available. The list was generated using information obtained from electrical drawings, the SQUG relay list, and the listing of basic events from the internal events PRA. It was then compared to the SQUG listing of low ruggedness relays to determine those at the plant that may remain for further evaluation under the SQUG program. Section A.2.4.4 contains additional details on this process.

Distributed Systems

Distributed systems reviewed in the walkdown included piping, cable trays, conduit, and HVAC ducting. General surveys of these systems to identify the presence of any seismic vulnerabilities were performed in walk-bys. Based on observations recorded in the walk-bys, specific samples of the distributed systems were selected for more detailed review to provide a basis for screening.

A.2.2.2.2 Walkdown Documentation

Documentation of the walkdown consisted of data sheets, photographs, and field notes for the equipment and structures surveyed. Walkdown data sheets following the formats recommended in EPRI NP-6041-SL were used. These data sheets vary according to the generic equipment component category. They contain checklists of seismic adequacy issues to be addressed in the inspection of a component and the data sheets include space to record additional notes and sketches. Photographs were taken to supplement any notes or details taken during the walkdown. For a component, photos of the overall configuration, anchorage, and any other notable features or systems interactions were typically taken.

A.2.2.3 Final Plant Walkdowns

Two subsequent plant walkdowns were performed. Walkdown of components inside the drywell was performed in March 1993 during the plant outage. In addition to walkdowns of the components inside the drywell that were included in the essential equipment list, a walk-by through the drywell was performed to identify any other potential sources of containment bypass, such as failure of containment penetrations due to systems interactions.

The final plant walkdown was performed in December 1993. Any components not previously surveyed were walked down at this time, including those in moderately radioactive areas that were previously inaccessible without specific radiation training. Several components previously surveyed were also re-inspected obtain additional detail required for seismic evaluation.

A.2.2.4 Findings from the Plant Walkdowns

Significant findings from the plant walkdowns are summarized below. Each of the potential seismic concerns discussed in Section A.2.5.

Structures. The civil structures were observed to be in general conformance with the design drawings. No gross deficiencies that could reduce seismic capacities were identified. Where accessible, the structure separation gaps were found to be as indicated on the design drawings.

Concrete Block Walls. Some of the concrete block walls targeted for walkdown were screened from further review based on a lack of failure consequences. Data on the remaining targeted walls were recorded for use in subsequent seismic evaluations.

Mechanical Equipment. The mechanical equipment was found to be seismically rugged and capable of retaining its structural integrity and post-earthquake functionality. Potential seismic concerns identified in the walkdowns included the following:

- The supports for service water automatic strainer F101 are not bolted to the anchors provided.
- Anchors for the reactor building component cooling water pumps may experience significant demands due to nozzle loads from piping with long unsupported spans.
- An adjacent, unanchored filter poses a system interaction concern to control rod drive pump P-201B.
- Vibration-isolated support legs for the room cooler adjacent to RHR pumps P-202B and 202D and core spray pump P-208B are relatively weak.
- Batteries for the start air compressors for diesel generators DG11 and DG12 are unrestrained.

Electrical Equipment. The electrical equipment was generally found to be adequate to retain its structural integrity and post-earthquake functionality. Anchorage upgrades added after original plant construction appeared to be substantial. Some potential seismic concerns that were identified in the walkdowns were associated with systems interactions and anchorage issues. Examples include the following:

- Relay panels C30 and C32 could be subjected to impact from adjacent panels or HVAC ducting. This is unlikely to affect component integrity, but could cause relay chatter.

- Battery chargers D52, D53, D54, D70, D80, and D90 are supported on channel sections that are subjected to weak-axis bending by side-to-side motion.

- Batteries for the diesel fire pump are unrestrained.

Tanks and Heat Exchanger. The tanks and heat exchanger themselves were generally found to be seismically adequate. Seismic concerns were generally confined to anchorage, including the following examples:

- The diesel day tanks were not welded to their supports during the initial plant walkdown. Welds were verified to be in place in the final walkdown.

- The condensate storage tanks are minimally anchored.

- The horizontally oriented diesel start air receivers are mounted to their supports by U-bolts. Some of the U-bolts were noted to be loose.

- The fuel oil supply tank for the diesel fire pump is supported by structural steel framing. The tank supports are covered with fireproofing, preventing confirmation of positive attachment of the supports to the framing.

Distributed Systems. The essential piping, cable trays, and conduit were found to be seismically adequate.

Other. The control room ceiling was found to be seismically vulnerable. The T-bar runners lack significant positive attachment to each other and could pull apart. The light fixtures are supported by the T-bars without independent wiring. The ceiling system lacks seismic bracing.

A.2.3 Seismic Response Analysis

The seismic response of the reactor, turbine, and control buildings were analyzed to obtain structure seismic loads and in-structure response spectra.

A.2.3.1 Reactor and Turbine Buildings

Conservative design in-structure response spectra that could also be used for USI A-46 resolution were generated for the reactor and turbine buildings. The calculated structure responses were later scaled up to earthquake ground motion corresponding to the conservative screening level finally selected. These analyses were performed to satisfy the requirements of the USNRC Standard Review Plan (SRP) [5]. The free-field ground motion consisted of a unique set of three time histories, two in the horizontal direction and one in the vertical direction. These response spectra match the regulatory guide R.G. 1.60 [6] ground spectra anchored to the SSE peak ground acceleration of 0.12g. A comparison of the generated spectra and the R.G. 1.60 spectra for one horizontal component is shown in Figure A.2.3-1.

Site response analyses were performed to obtain the soil material properties consistent with the seismic-induced strains. Low strain soil properties were obtained from analyses by Harding-Lawson Associates [7]. Low strain soil shear wave velocities considered in the site response analysis varied from about 950 ft/sec at grade to 2,300 ft/sec at a depth of 100 feet. Three soil profiles were generated by the site response analysis: Best estimate, lower bound, and upper bound. Low strain soil shear moduli for the lower and upper bound profiles were taken to be one-half and twice those for the best estimate profile. Reductions in soil stiffness and increases in soil material damping with increased strain were based on relationships presented by Seed and Idriss [8]. The strain-compatible soil properties were generated using the SHAKE computer program [9].

The foundation impedance and scattering functions were calculated for the building-specific foundation configurations. The SASSI computer program SASSI [10], which can account for non-circular embedded foundations, was used for the reactor building. The turbine building was modeled as being surface-founded using the CLASSI computer program [11]. Strain-compatible soil properties obtained from the site response analyses were used in these calculations.

Existing structure dynamic models were used, with minor modifications. The reactor building model, including the drywell and the reactor pressure vessel, was based on the structure model used in the original design seismic analysis [12]. The turbine building model was developed from an analysis by NUTECH Engineers [13].

In-structure response spectra were generated for the best estimate, lower bound, and upper bound soil cases. Spectra from the three cases were enveloped. Representative examples of conservative design in-structure response spectra for the reactor and turbine buildings are shown in Figures A.2.3-2 through A.2.3-5. The response reductions associated with the soil-structure interaction are evident.

A.2.3.2 Control Building

The seismic response of the control building was analyzed in a manner similar to the reactor and turbine building analyses, with the following exceptions:

- Free-field ground motion input consisting of three ground motion time histories, two horizontal and one vertical, whose spectra match a median NUREG/CR-0098 [14] shape anchored to a peak ground acceleration of 0.3g were generated (Figure A.2.3-6).

- Three soil profiles were considered: Best-estimate, lower bound, and upper bound. The high strain shear moduli for the lower and upper bound profiles were taken to be 0.5 and 1.5 times the high strain best-estimate moduli.

Representative examples of in-structure response spectra for the control building are shown in Figures A.2.3-7 and A.2.3-8. The soil-structure interaction significantly reduces the response spectra of the reactor and turbine buildings, but this is effect is not apparent in the control building spectra. The fundamental soil-structure system frequencies vary with the soil case and are between 3 to 7 Hz.

Figure A.2.3-1 Comparison of Synthetic Time History Response to R.G. 1.60 Target Spectrum: Horizontal Time History

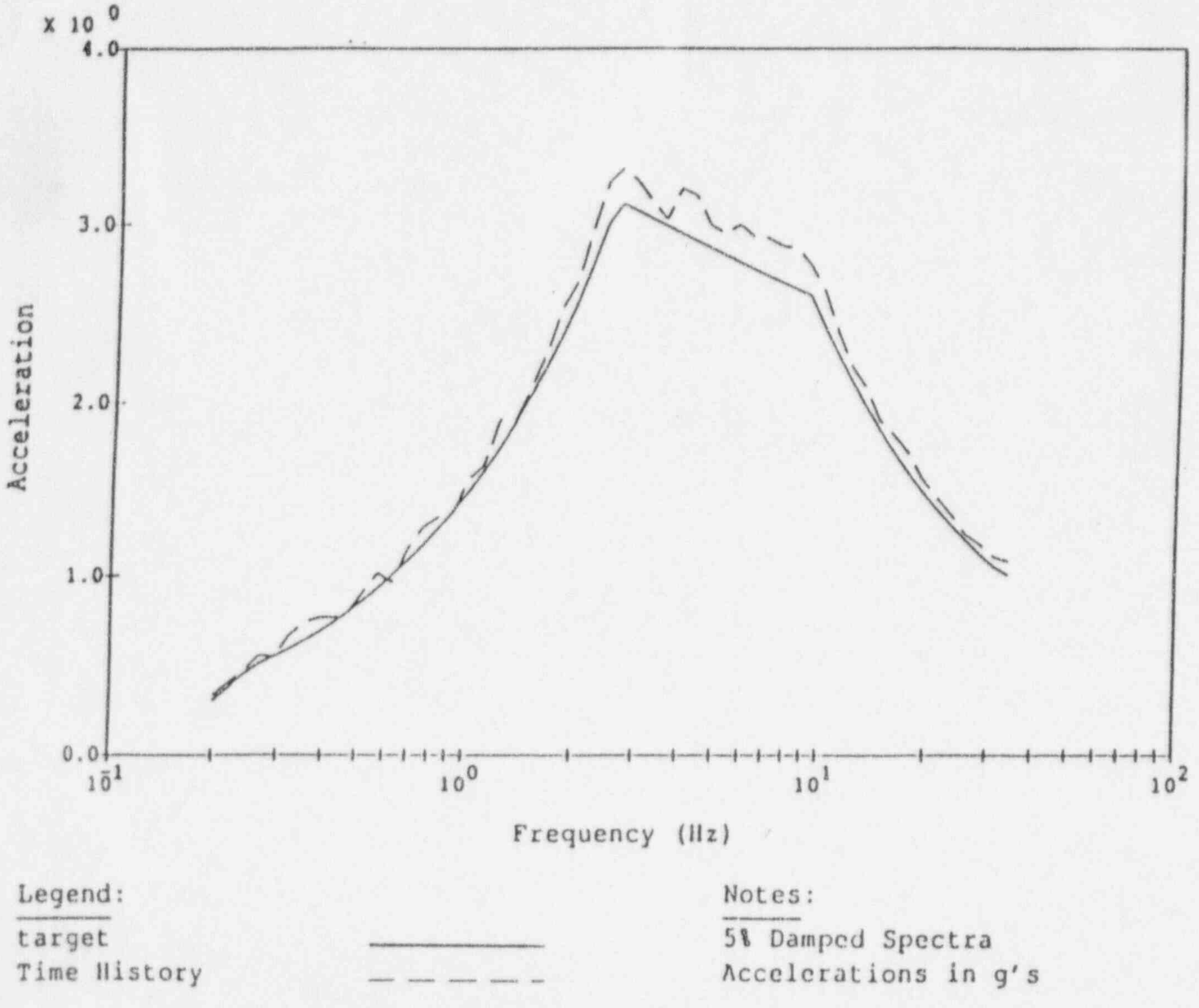
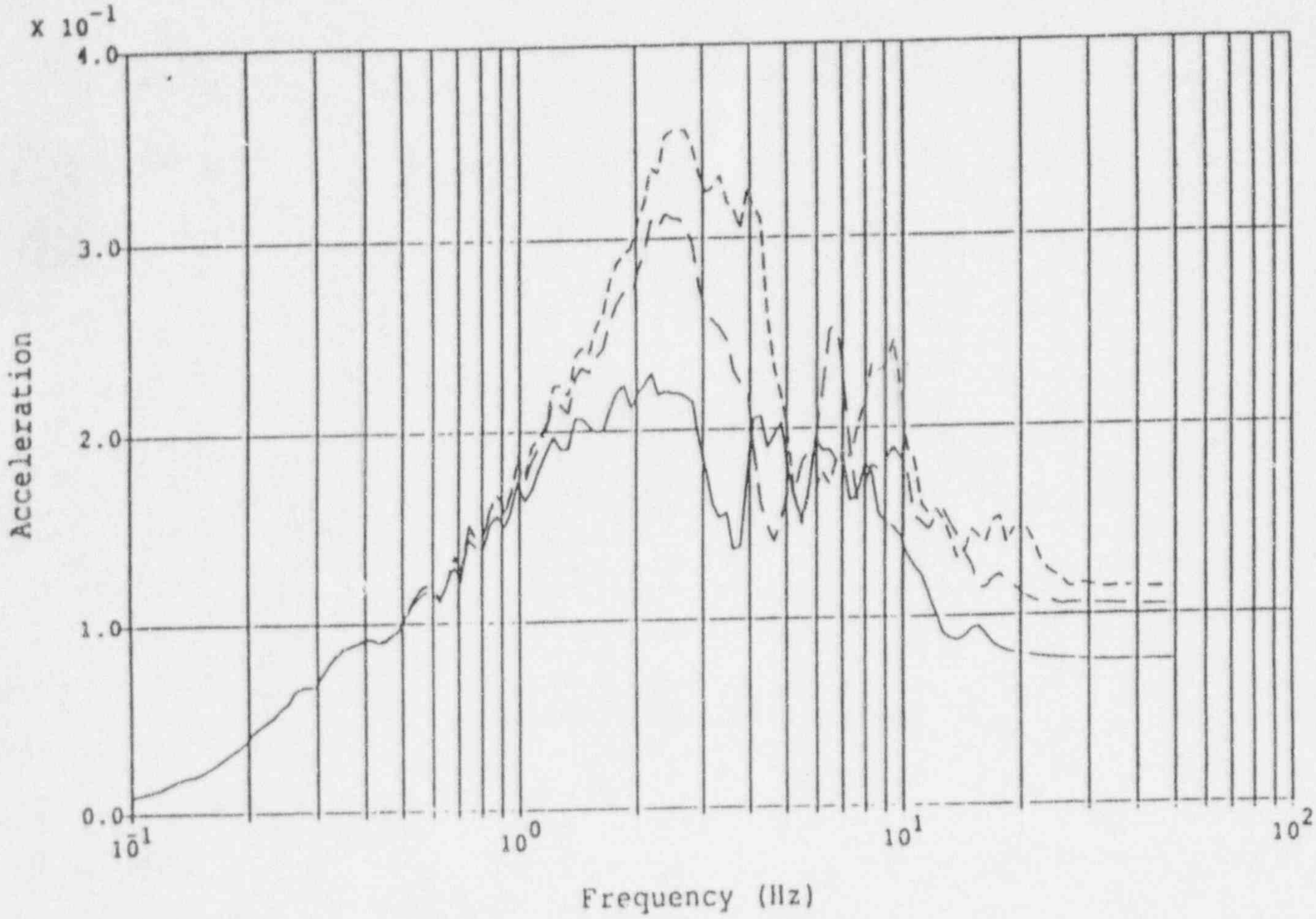


Figure A.2.3-2 Monticello Reactor Building, R.G. 1.60 Deterministic Run, Foundation Node 28, Elev. 896'-3", Translation in NS Direction



Legend:

Lower Bound Soil Properties ————————

Best Estimate Soil Properties — — — — —

Upper Bound Soil Properties ---------

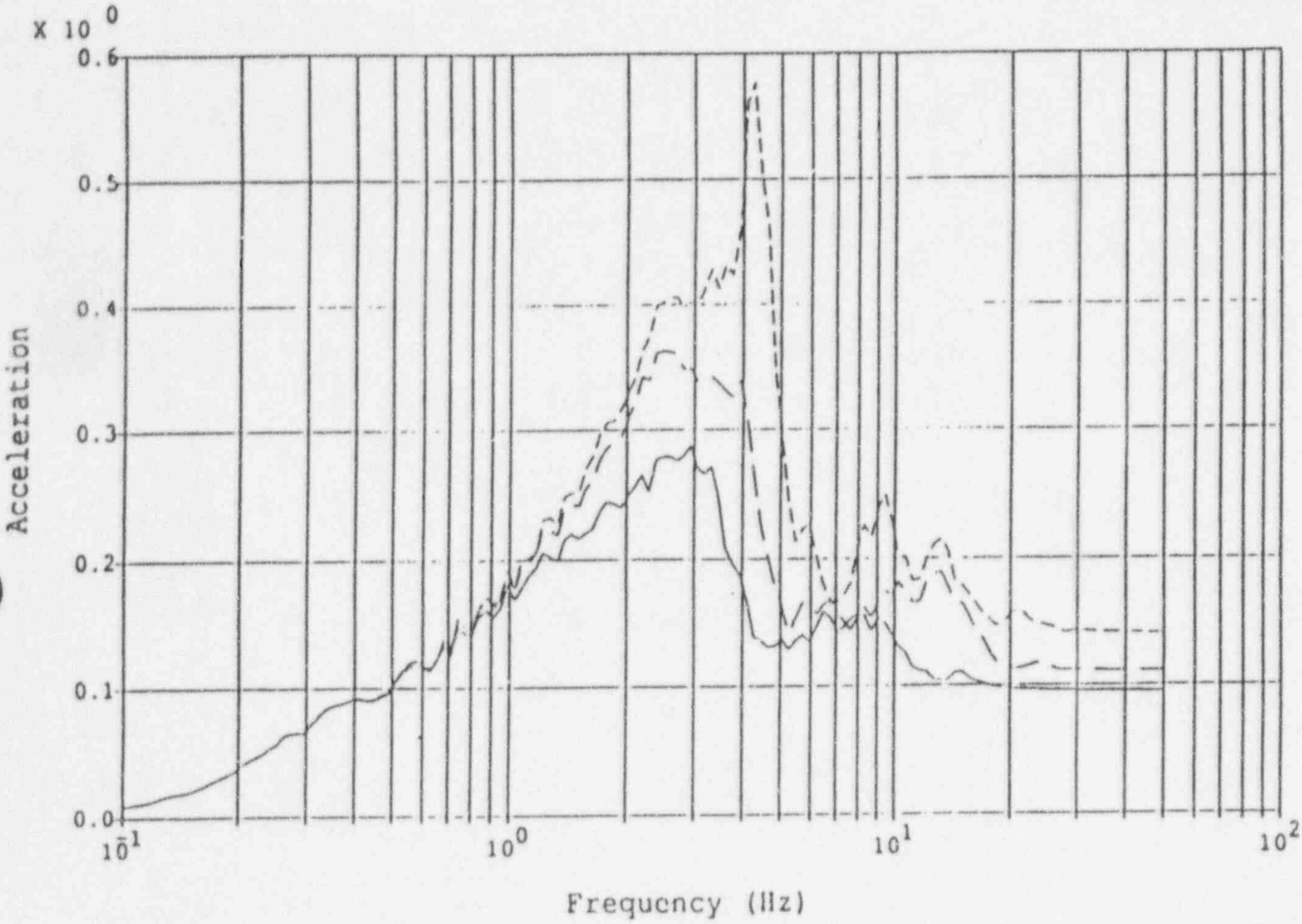
Notes:

5.0 % Spec. Damping

Accelerations in g's

1 SSE Level = 0.12g

Figure A.2.3-3 Monticello Reactor Building, R.G. 1.60 Deterministic Run, Drywell Stick Node 01, Elev. 935'-0", Translation in NS Direction



Legend:

Lower Bound Soil Properties ————————

Best Estimate Soil Properties — — — — —

Upper Bound Soil Properties - - - - - - - - -

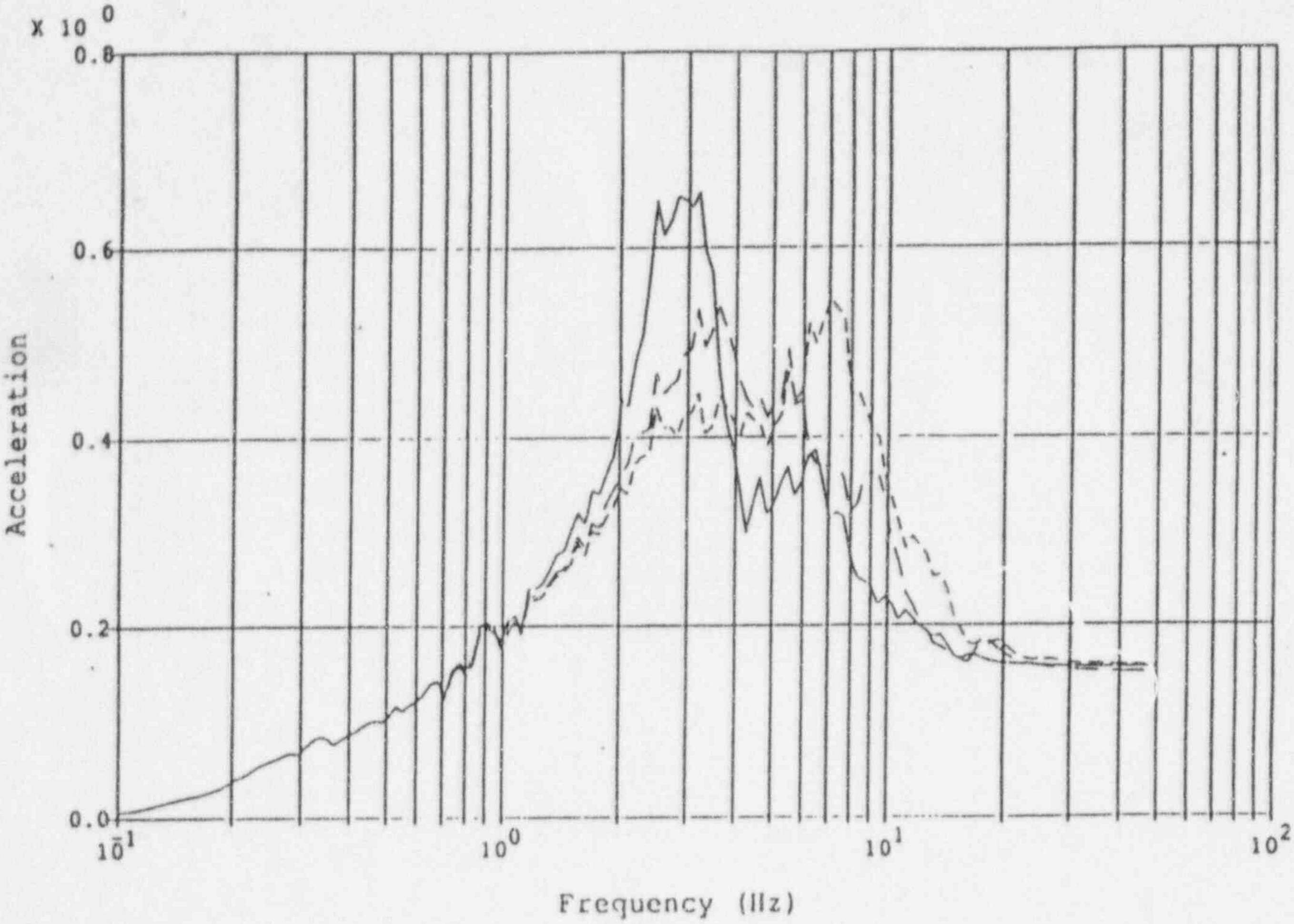
Notes:

5.0 % Spec. Damping

Accelerations in g's

1 SSE Level = 0.12g

Figure A.2.3-4 Monticello Turbine Building, R.G. 1.60 Deterministic Run, Foundation Node 01, Elev. 911'-0", Translation in NS Direction



Legend:

Lower Bound Soil Properties

Best Estimate Soil Properties

Upper Bound Soil Properties

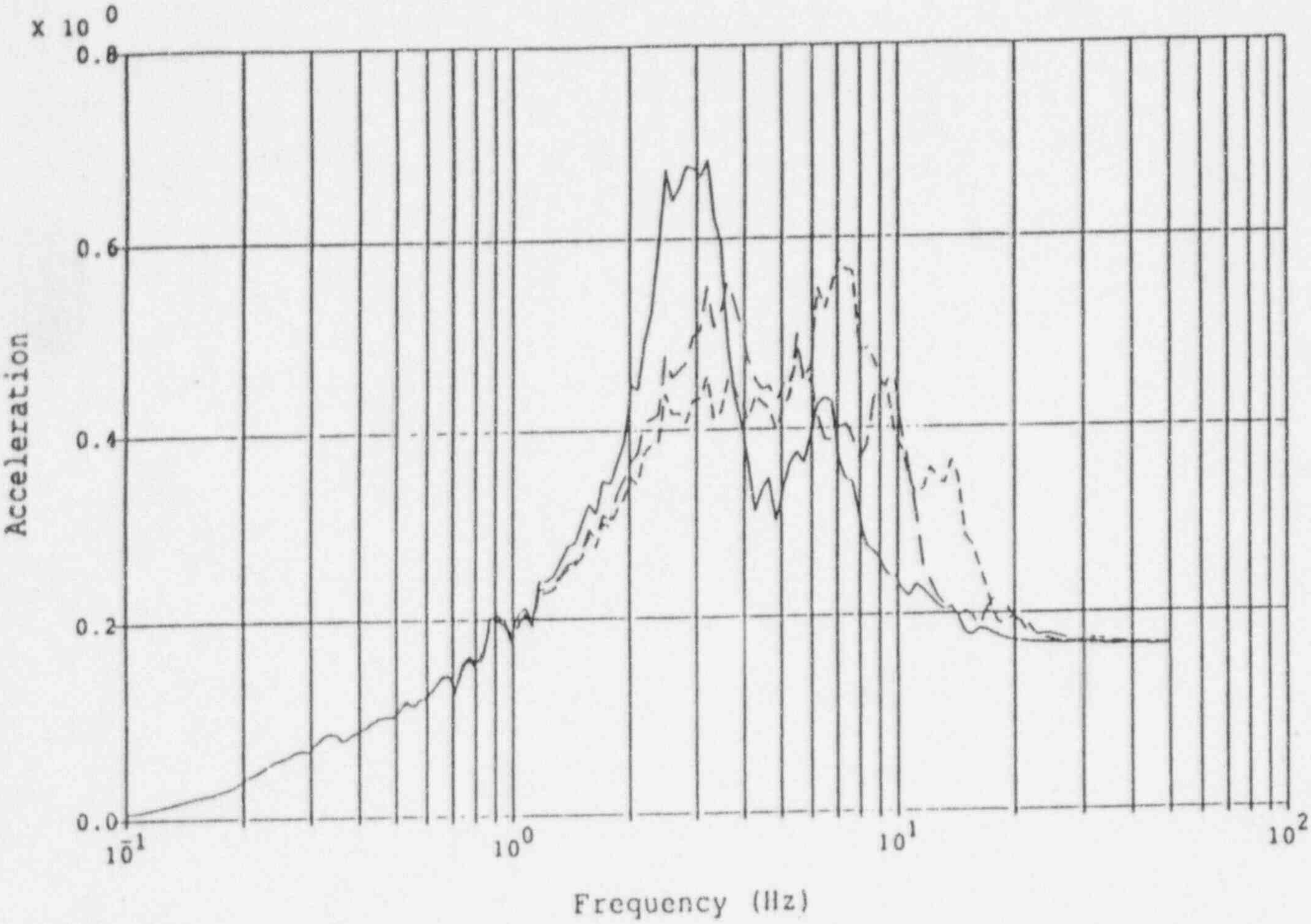
Notes:

[illegible] % Spec. Damping

Accelerations in g's

1 SSE Level = 0.12g

Figure A.2.3-5 Monticello Turbine Building, R.G. 1.60 Deterministic Run, Main Structure Node 04, Elev. 931'-0", Translation in NS Direction



Legend:

Lower Bound Soil Properties ————————

Best Estimate Soil Properties — — — — -

Upper Bound Soil Properties - - - - - - - - -

Notes:

5.0 % Spec. Damping

Accelerations in g's

1 SSE Level = 0.12g

Figure A.2.3-6 Freefield Ground Response Spectrum Matching Median NUREG/CR-0098, Monticello Control Building

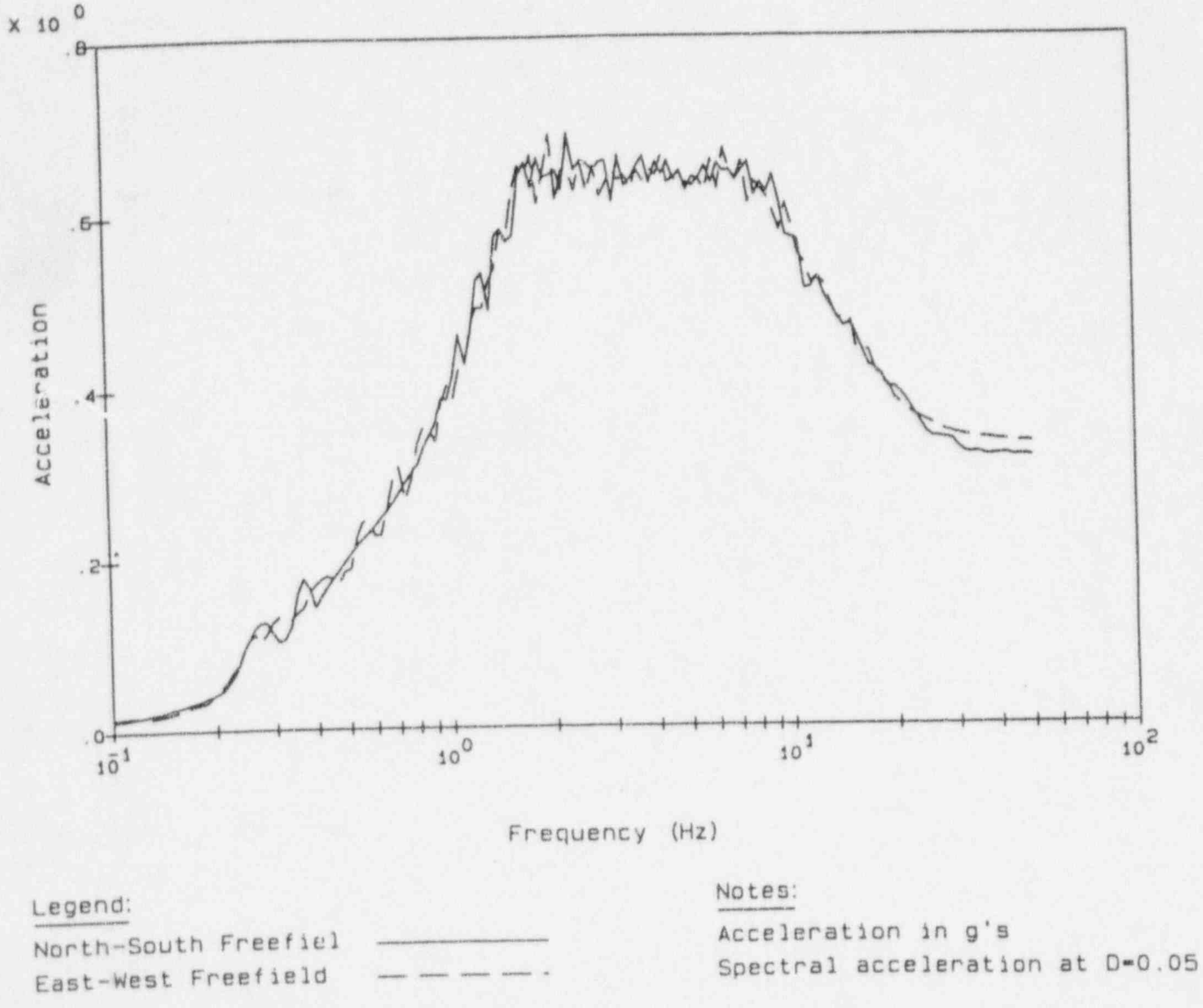
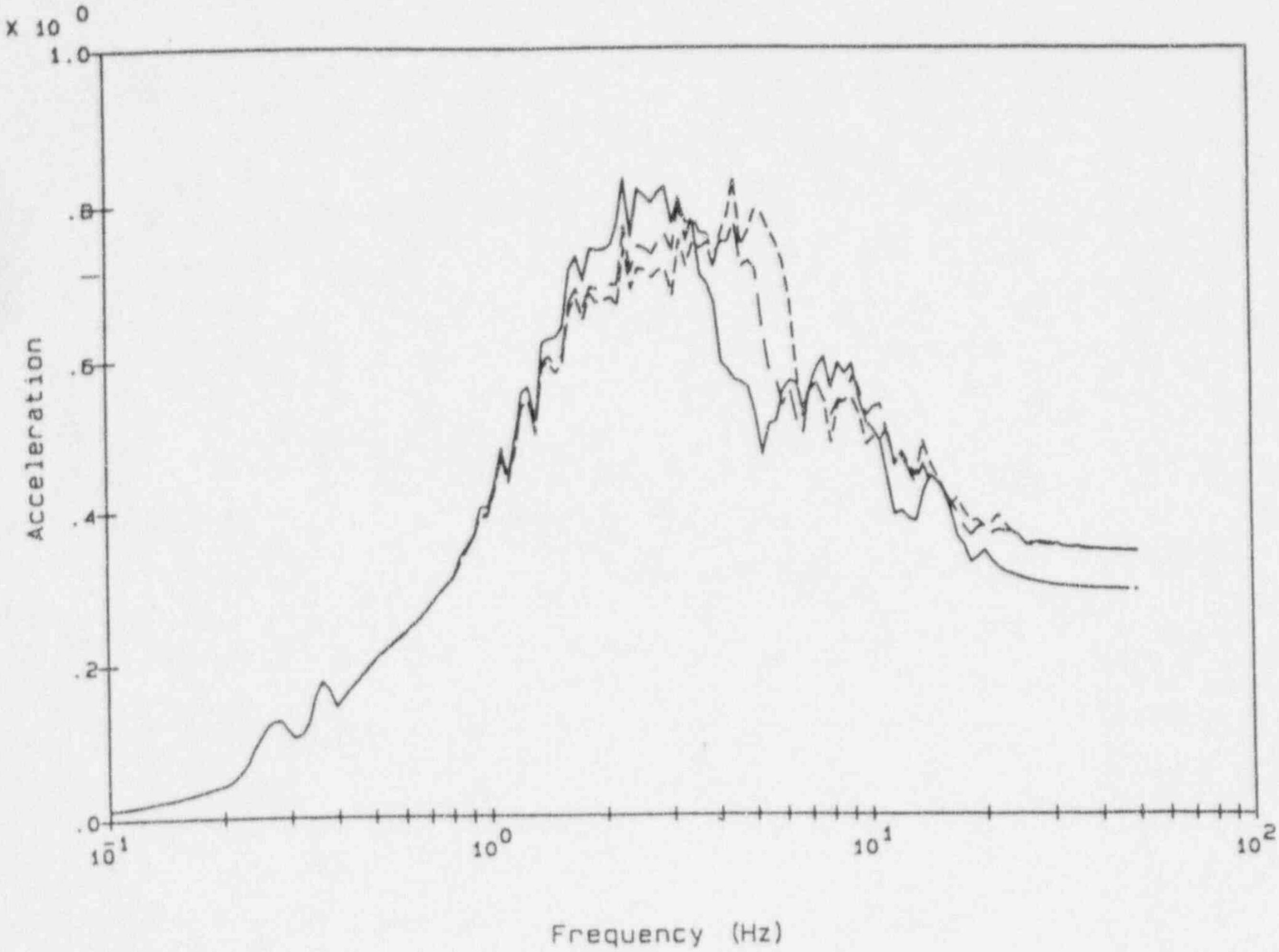


Figure A.2.3-7 Representative In-Structure Response Spectrum for Monticello Control Building, Elev. 928'-0"



Legend:

Found. Lower Bound ————————

Found. Best Estimate — — — — -

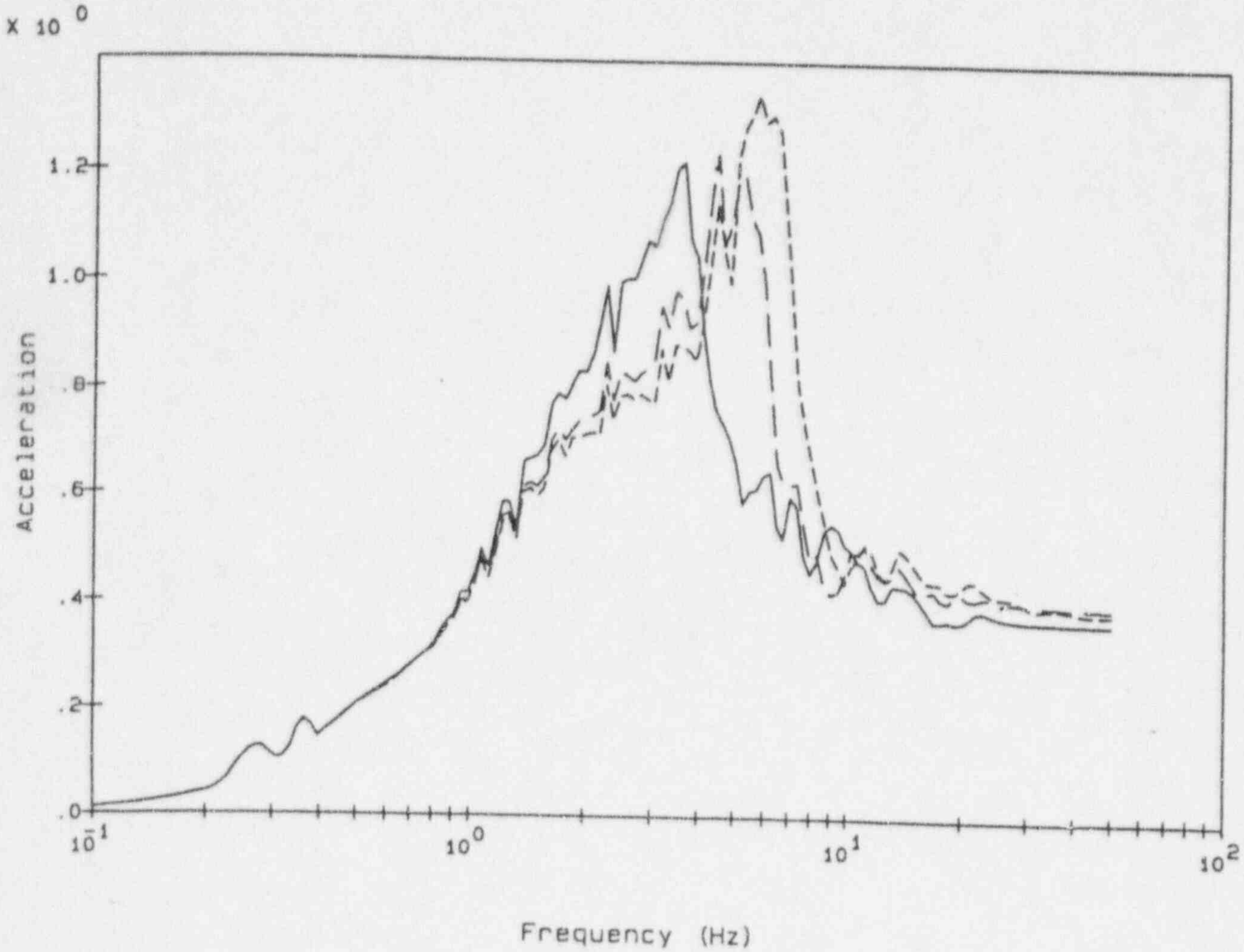
Found. Upper Bound ---------

Notes:

Acceleration in g's

Spectral acceleration at D=0.05

Figure A.2.3-8 Representative In-Structure Response Spectrum for Monticello Control Building, Elev. 951'-0"



Legend:

Lower Bound ————

Best Estimate — — — —

Upper Bound - - - - - - -

Notes:

Acceleration in g's

Spectral acceleration at D=0.05

A.2.4 Component Screening

Components having relatively high seismic capacities were determined to need no further review following the methodology for seismic margin assessment of EPRI NP-6041-SL. This screening was performed for a seismic margin earthquake having a peak 5 percent damped spectral acceleration of 0.8g or less. This screening level was considered to be conservative for a reduced scope seismic margins assessment of Monticello.

The criteria for screening equipment from further review are summarized in the left column in Tables 2-3 and 2-4 of EPRI NP-6041-SL. This screening was based on walkdowns (Section A.2.2) supplemented by quantitative evaluations as required. The screening evaluations were typically based on (1) the structure and in-structure seismic responses from the seismic response analyses described in Section A.2.3, scaled to the seismic margin earthquake level and (2) the acceptance criteria recommended by EPRI NP-6041-SL. These typically were bounding calculations which used conservative, simple approximations and input in order to screen components from more detailed analysis.

A.2.4.1 Structure Screening

The following structures were found not to need more detailed analysis:

- Reactor building
- The portions of the turbine building which house Class I equipment
- Control building
- Intake structure
- Emergency filtration train building
- Diesel generator building
- Drywell
- Suppression chamber
- Class II buildings

The reactor building, portions of the turbine building which house Class I equipment, the control building, intake structure, EFT building, and diesel generator building are all categorized as Class I structures in the USAR. (The turbine building itself was designed as a Class II structure, but those portions which house Class I equipment were analyzed for Class I loads and are included here with the Class I structures.) The seismic load-resisting systems of these structures consist of reinforced concrete shear walls, floor diaphragms, and foundations designed for an SSE of 0.12g peak ground acceleration. These structures satisfy the screening criteria. To supplement the screening, building drawings were reviewed to identify any significant seismic vulnerabilities. Conservative bounding calculations of selected structural elements verified that the Class I structures can be screened from further analysis.

The drywell satisfies the screening criteria applicable to free-standing steel containments. It was designed to withstand the combined loads due to the SSE (obtained by dynamic analysis) and a concurrent loss of coolant accident. The support skirts for the drywell and the internal structure lining are anchored to the reactor building base mat and embedded in the drywell foundation concrete.

A detailed analysis of the Monticello primary containment system was performed as documented in the Plant Unique Analysis Report (PUAR) [15]. A review of the analysis indicates that, with the exception of the suppression chamber seismic restraints, the stresses in the containment system due to seismic loads are small in comparison to those due to other loads. EPRI NP-6041-SL requires an evaluation of Mark I suppression chambers for earthquakes which exceed the design basis. Suppression chamber responses for the seismic margins screening were obtained by scaling the SSE responses reported in Volume 2 of the PUAR. The seismic loads on the suppression chamber seismic restraints were combined with the loads due to SRV discharge and the hydrodynamic chugging which may occur at the downcomers from the drywell to the pool under certain small LOCA conditions. The capacities of the most highly stressed seismic restraint components, based on the EPRI NP-6041-SL acceptance criteria, were found to exceed these applied loads. The suppression chamber therefore did not need a detailed fragility evaluation.

Bounding calculations were performed to investigate the potential for structure-to-structure impact between adjacent Class I structures. These calculations verified that the existing building separations are sufficient to prevent impact, and therefore structure-to-structure impact need not be considered further.

Class II structures which are in the vicinity of the Class I structures include the administration building, radwaste building, those portions of the turbine building which do not house Class I equipment, the turbine building addition, and the boiler building. These structures were designed to meet the Seismic Zone 1 requirements of the 1964 Uniform Building Code. A review of the structural drawings showed that even if a Class II structure were to fail and impact an adjacent Class I structure, the Class I structure and essential equipment would be unlikely to fail. The complete collapse of steel-framed Class II structures onto Class I structures below them was judged to be unlikely except during earthquakes above the screening basis.

A.2.4.2 Concrete Block Wall Screening

EPRI NP-6041-SL requires seismic evaluation of masonry walls for all earthquake levels. As discussed in Sections A.2.2.2.1 and A.2.2.4, during the walkdown some of the block walls were found to need no further review because their failure would have no significant consequences. Parametric comparisons of the remaining walls were performed to select bounding case walls for detailed evaluation. The selected walls included the following:

Wall	Building
Wall 243	Reactor Building
Walls C105 and C110	Control Building
Wall D105	Diesel Generator Building
Wall T109	Turbine Building
Diesel fire pumphouse east wall	Intake Structure

The acceptance criteria for unreinforced concrete block walls were based on the allowable stresses specified by ACI 530-88/ASCE 5-88 [16]. Reinforced concrete block walls were evaluated following the procedures recommended in Appendix R of EPRI NP-6041-SL. The bounding case walls were found to have sufficient seismic capacity, and therefore all the remaining concrete block walls were screened from further review.

A.2.4.3 Component Screening

The components that were found to need no further review are listed in Table A.2.4-1, and some general comments on these components are summarized below. The remaining components, for which further analysis was needed, are discussed in greater detail in Section A.2.5.

Mechanical Equipment

All mechanical equipment was verified to be adequate to retain its structural integrity and post-earthquake functionality based on the walkdowns. System interaction concerns associated with failure of adjacent components onto some of the RHR and core spray pumps were identified in the walkdown (Section A.2.2.4). These interactions were flagged for further review.

Anchorage was evaluated using bounding cases for the different generic component categories. For example, many of the horizontal pumps were represented by the evaluation of a single bounding case. When the bounding case was found to satisfy the EPRI NP-6041-SL criteria, all of the individual components of that generic category were also considered to meet the criteria and need no further review.

Electrical Equipment

All electrical equipment was verified to be adequate to retain its structural integrity and post-earthquake functionality based on the walkdowns. System interaction concerns associated with impact between essential electrical cabinets and adjacent components were identified in the walkdown (Section

A.2.2.4). While the structural integrity of the essential cabinet would not necessarily be threatened, there is a potential for chatter of essential relays due to impact. These interactions were flagged for further review.

The anchorage of electrical equipment was evaluated following the EPRI NP-6041-SL criteria. The anchorages of most essential electrical components which were installed as part of the original plant construction were previously evaluated for design basis seismic loads [17]. The anchorage and supports for many components were subsequently upgraded based on that study. Screening calculations for the IPEEE were consequently performed by scaling the design basis seismic evaluation/design results to the EPRI NP-6041-SL criteria. Screening evaluations were also performed for other essential electrical equipment which was not included in the design basis evaluations noted above.

Those components whose seismic capacities did not meet these screening criteria and components having specific anchorage concerns identified in the walkdown were flagged for further review.

Tanks and Heat Exchanger

With a couple of exceptions, the tanks and heat exchanger were found to need no additional review based on the walkdowns and bounding calculations. For example, many of the vertical tanks were aggregated into a single group. By analyzing the anchorage of a selected bounding case, the anchorages of all of these tanks were determined to need no further review.

Distributed Systems

A walkdown of representative systems, supplemented by a review of the drawings, verified that the essential piping is seismically adequate and needed no further review. All valves were screened from further review by data gathered during the walkdowns. Cable trays and conduit were also found to have sufficiently high seismic capacities based on a walkdown of representative examples.

HVAC ducting is not required for room cooling. It was reviewed only as a potential source of system interactions with essential equipment. Any HVAC ducting in close proximity to essential components was flagged for further review during the walkdowns. The ducting is typically supported by metal straps anchored to the overhead concrete floor slab by shotpins. A conservative bounding calculation verified that the shotpins have sufficient pullout capacity. HVAC ducting therefore needed no further review.

A.2.4.4 Relay Screening

Because Monticello is included in the SQUG program, the evaluation of relay chatter at Monticello was conducted following SQUG procedures. From the SQUG relay list it was determined that Monticello does have some relays which are considered to have low seismic ruggedness. The evaluation of relay chatter was therefore expanded to include relays outside of the scope of the SQUG program but within the scope of the IPEEE (Section 3.2.4.2 of NUREG-1407). An initial identification and functional screening of the relays was done in order to acquire an understanding of the potential effect of relay chatter.

The evaluation for the seismic IPEEE proceeded by generating a list of relays appropriate for the IPEEE models. Also in the list, where available, is information concerning the relay's configuration, location, manufacturer, and model number. This relay list and its information was gathered from the Monticello electrical drawings, the SQUG relay list, and the list of basic events from the Monticello internal events PRA. Thus, the list contains all relays appropriate to either the SQUG program or to the IPEEE. This list was then compared to the "bad actors" list in EPRI NP-7148 [18] to determine if there were any relays with low seismic ruggedness (as defined for the SQUG program) within the plant, regardless of whether they are within the scope of the SQUG program.

Table A.2.4-2 contains the list of all the relays at Monticello which have a manufacturer and model number specified in the list of low-ruggedness relays ("bad actors") in Appendix E of EPRI NP-7148. If a relay's manufacturer or model number could not be obtained, the relay was included in the table. However, for a low-ruggedness relay to be of concern, it must also operate in a configuration that makes it susceptible to chatter, as listed in EPRI NP-7148. Table A.2.4-2 includes all the potential low-ruggedness relays, regardless of their operating configurations.

As indicated by the information in the table, all the potential low-ruggedness relays at Monticello for which the manufacturer and model type could be determined are of the GE/HGA model. For these relays, the configuration of concern is deenergized, normally closed (DE, NC); if the configuration of the relay is different form this, then it need not be considered further. The actual configurations of the relays are listed in Table A.2.4-2. If the configuration could not be determined, the relay was assumed to be in the DE, NC mode, and therefore a potential "bad actor".

A review of Table A.2.4-2 shows that several of the relays listed are in fact not of concern because their operating configurations do not make them susceptible. For those relays known or assumed to be in the susceptible DE, NC configuration, a functional analysis was conducted to determine the consequences if the relay failed to operate. Table A.2.4-2 gives that information in the columns labeled "Function" and "Comments". For several of the relays, failure of the relay has no effect on the plant's response to a seismic event, either because its failure does not prevent the system from operating or because the system already cannot operate since the seismic event is assumed to cause a loss of offsite power.

Through these various screening evaluations, the original list of potential "bad actor" relays shown in Table A.2.4-2 was eventually pared down to only the few which are summarized in Table A.2.4-3. These relays are all in the RHR system, and may affect its initiation or operation. All of these relays are within the scope of the SQUG program; no relay outside the SQUG program was found to be a "bad actor."

Table A.2.4-1 Seismically Rugged Components

Component	ID No.	System
Diesel generator	DG11	AC power
Diesel generator	DG12	AC power
DG11 day tank	T45A	AC power
DG12 day tank	T45B	AC power
Diesel oil storage tank	T44	AC power
Diesel 11 air start air compressors (2 total)	K-8A/8B	AC power
Diesel 12 air start air compressors (2 total)	K-9A/9B	AC power
DG 11 room fan	VSF-9	AC power
DG 12 room fan	VSF-10	AC power
DG 11 relay panel	C91	AC power
DG 12 relay panel	C92	AC power
DG 11 control panel	C93	AC power
DG 12 control panel	C94	AC power
DG 11 neutral ground transformer	G31	AC power
DG 12 neutral ground transformer	G41	AC power
Air intake/exhaust louvers		AC power
Lube oil pumps (2 total)		AC power
Distribution panel	Y10	AC power
Distribution panel	Y20	AC power
Distribution panel	Y30	AC power
UPS 120V AC panel	Y70	AC power
UPS 120V AC panel	Y80	AC power
4KV bus	13	AC power
4KV bus	15	AC power
480 bus	LC-103	AC power
480 bus	LC-104	AC power
Motor control center	MCC-33A	AC power
Motor control center	MCC-33B	AC power
Motor control center	MCC-34	AC power
Motor control center	MCC-44	AC power
4KV to 480V transformer	TRX30	AC power
4KV to 480V transformer	TRX40	AC power

Table A.2.4-1 (continued) Seismically Rugged Components

Component	ID No.	System
Bus transfer	#12	AC power
Switch panel	Y73	AC power
Manual bypass switch	Y83	AC power
Inverter panel	Y71	AC power
Inverter panel	Y81	AC power
Main control panel	C08	AC power
ATWS Channel A/ B	9-95/96	ATWS
Main control panel	C03	Automatic Depressurization
Safety relief valves (8 total)	2-71A to H	Automatic Depressurization
Tail pipes to torus		Automatic Depressurization
Accumulators for SRVs (8 total)	T-57A to H	Automatic Depressurization
Feedwater check valve	FW-67-1	Condensate and feedwater
Feedwater check valves (2 total)	94-1/-2	Condensate and feedwater
Feedwater check valves (2 total)	97-1/-2	Condensate and feedwater
CRD pump	P-201A	Control rod drive
CRD hydraulic control unit accumulators		Control rod drive
Scram discharge volume tank	SDV-A	Control rod drive
Scram discharge volume tank	SDV-B	Control rod drive
CRD flow control valves (2 total)	CV 3-19A/B	Control rod drive
CRD suction valve from CST	CRD-1	Control rod drive
Press. control valve from cond. reject line	PCV 3-23	Control rod drive
Check valve from cond. reject line	CRD-67	Control rod drive
Check valve from CSTs	CRD-69	Control rod drive
Press. control valves from drive water	MO3-20/-22	Control rod drive
Check valves in lines to RPV (2 total)	CRD-4-1/-2	Control rod drive
Water filters in lines to RPV (2 total)	F-200A/200B	Control rod drive
CS injection valve	MO-1753	Core spray
CS injection valve	MO-1754	Core spray
Torus suction valve	MO-1741	Core spray
Torus suction valve	MO-1742	Core spray
Relay Panel C33	C33	Core Spray

Table A.2.4-1 (continued) Seismically Rugged Components

Component	ID No.	System
125V DC Panel D11	D11	DC power
125V DC Panel D21	D21	DC power
125V DC Panel D33	D33	DC power
125V DC Panel D111	D111	DC power
125V DC Panel D211	D211	DC power
250V DC Panel D100	D100	DC power
250V DC Panel D31	D31	DC power
250V DC battery	D3	DC power
250V DC battery	D6	DC power
125V DC battery	D11	DC power
125V DC battery	D12	DC power
Battery chargers	D10	DC power
Battery chargers	D20	DC power
Battery chargers	D40	DC power
ECCS instrument rack	C55	ECCS
ECCS instrument rack	C56	ECCS
Relay panel	C303A	ECCS
Relay panel	C303B	ECCS
EDG-ESW pumps (2 total)	P-111A/B	EDG-ESW
Check valves (2 total)	ESW-1-1/2	EDG-ESW
ESW strainers (2 total)	BS-1980/ 2414	EDG-ESW
Diesel fire pump	P-105	Fire protection
Diesel fire pump control panel	C-104	Fire protection
RHR/Fire water crosstie valve	RHRSW 46	Fire protection
HPCI pump/turbine	P-209	HPCI
HPCI aux oil pump	P-217	HPCI
HPCI gland seal condenser	E-204	HPCI
Turbine stop valve	HO-7	HPCI
Steam to turbine isolation valve	MO-2036	HPCI
Suppression pool suction valves	MO-2061/ 2062	HPCI
HPCI injection valve	MO-2067	HPCI

Table A.2.4-1 (continued) Seismically Rugged Components

Component	ID No.	System
HPCI injection valve	MO-2068	HPCI
CST pump suction valve	MO-2063	HPCI
Relay panel	C39	HPCI
HPCI instrument rack	C-120	HPCI
N2 bottles for SRVs		Instrument air
Main steam isolation valves (inboard)	A0-2-80A to D	Main condenser
Main steam isolation valves (outboard)	A0-2-86A to D	Main condenser
MSIV accumulators (inboard)	T-49A to -D	Main condenser
MSIV accumulators (outboard	T-50A to D	Main condenser
RBCCW surge tank	T-3	RBCCW
RCIC pump/turbine	P-207	Reactor core isolation cooling
RCIC barometric condenser	E-203	Reactor core isolation cooling
RCIC valve	MO-2078	Reactor core isolation cooling
RCIC valve	MO-2096	Reactor core isolation cooling
RCIC valve	MO-2100	Reactor core isolation cooling
RCIC valve	MO-2101	Reactor core isolation cooling
RCIC valve	MO-2106	Reactor core isolation cooling
RCIC valve	MO-2107	Reactor core isolation cooling
RCIC valve	MO-2102	Reactor core isolation cooling
Relief valve	RV-2097	Reactor core isolation cooling
RCIC instrument rack	C-128	Reactor core isolation cooling
Scram pilot valves (2 total)	SO-117/ 118	Reactor protection system
Backup scram valves (2 total)	SV-3-140A/ B	Reactor protection system
SDV vent and drain pilot valves	SV-3-31A to D	Reactor protection system
SDV vent and drain pilot valves	SV-3-33A to D	Reactor protection system
RHR injection check valves	AO-1046A/B	Residual heat removal
RHR injection valves	MO-2014/2015	Residual heat removal
Loop suction valve from torus	MO-1986	Residual heat removal
Loop suction valve from torus	MO-1987	Residual heat removal
Loop discharge valve to torus	MO-2006	Residual heat removal
Loop discharge valve to torus	MO-2007	Residual heat removal

Table A.2.4-1 (continued) Seismically Rugged Components

Component	ID No.	System
Torus cooling injection valves	MO-2008/2009	Residual heat removal
Cont. spray inboard isol. valve	MO-2022	Residual heat removal
Cont. spray inboard isol. valve	MO-2023	Residual heat removal
Cont. spray outboard isol. valve	MO-2020	Residual heat removal
Cont. spray outboard isol. valve	MO-2021	Residual heat removal
RHR heat exchanger	E-200A	Residual heat removal
RHR heat exchanger	E-200B	Residual heat removal
RHRSW pumps	P-109A to D	RHR service water
Control valve from HX discharge	CV-1728	RHR service water
Control valve from HX discharge	CV-1729	RHR service water
Relief valves (2 total)	RV-3039/3038	RHR service water
RHRSW to RHR Crosstie Valve	RHRSW-14	RHR service water
Service water pumps	P-102A to C	Service water
SW discharge check valves	SW-1-1 to -3	Service water
SW valves to condensate system	SW-145/ 146	Service water
SW/condensate HW emergency fill valve	SW-147	Service water
SLC pumps	P-203A/ B	Standby liquid control
SLC pump discharge check valves	XP-3-1/ 2	Standby liquid control
SLC injection check valve	XP-6	Standby liquid control
SLC injection check valve	XP-7	Standby liquid control
SLC explosive valves	11-14A/ B	Standby liquid control
Main control panel	C05	Standby liquid control
SLC tank	T-200	Standby liquid control
Vacuum breakers	AO-2382A-K	Vapor suppression
Tail Pipes to torus		Vapor suppression
Reactor Pressure Vessel		
Piping and piping penetrations		
HVAC ducting		
Cable trays and conduit		

Table A.2.4-2 Initial List of Potential Low Ruggedness Relays

Relay	Manufacturer/ Type	Function	Relay[a] Configuration	Comments
AC Power Relays				
102-5	G.E./HGA	152-502	DE, AC, NO[b]	
102-6	G.E./HGA	152-602	DE, AC, NO[b]	
127-DG1	G.E./HGA	152-502	DE, AC, NO[b]	
127-DG1X	G.E./HGA	152-502	DE, AC, NO[b]	
127-DG2	G.E./HGA	152-602	DE, AC, NO[b]	
127-DG2X	G.E./HGA	152-602	DE, AC, NO[b]	
186ST		Breaker 5N5 control		Offsite power assumed unavailable.[d]
1AR L/O		4KV Bus 1AR lockout		Offsite power assumed unavailable.[d]
86-101A		Transformer 10 TR protection		Offsite power assumed unavailable.[d]
86-101B		Transformer 10 TR protection		Offsite power assumed unavailable.[d]
86-102A		Transformer 10 TR protection		Offsite power assumed unavailable.[d]
86-102B		Transformer 10 TR protection		Offsite power assumed unavailable.[d]
86-61XA		Transformer 1R protection		Offsite power assumed unavailable.[d]
86-61XB		Transformer 1R protection		Offsite power assumed unavailable.[d]
86-62XA		Transformer 1R protection		Offsite power assumed unavailable.[d]
86-62XB		Transformer 1R protection		Offsite power assumed unavailable.[d]
Core Spray Relays				
14A-K23A/B	G.E./HGA	Pump running signal to ADS	DE, DC, NO[b]	
14A-K25A/B	G.E./HGA	Pump running signal to ADS	DE, DC, NO[b]	

Table A.2.4-2 (continued) Initial List of Potential Low-Ruggedness Relays

Relay	Manufacturer/ Type	Function	Relay[a] Configuration	Comments
HPCI Relays				
23A-K5	G.E./HGA		DE, DC, NO[b]	
23A-K6	G.E./HGA		DE, DC, NO[b]	
23A-K7	G.E./HGA	Auto closure of drain valves	DE, DC, NC	Will not impact system operation[c]
23A-K8	G.E./HGA		DE, DC, NO[b]	
23A-K14	G.E./HGA		DE, DC, NO[b]	
23A-K16	G.E./HGA	Pump discharge low flow	DE, DC, NC	Would only result in closure of minimum flow valve, which is not required for system operation[c]
23A-K17	G.E./HGA		DE, DC, NO[b]	
23A-K20	G.E./HGA		DE, DC, NO[b]	
23A-K24	G.E./HGA		DE, DC, NO[b]	
23A-K30	G.E./HGA	Indicating lights for testable check valve	DE, AC	Indication only[c]
23A-K31	G.E./HGA		EN, DC, NC	Indication only[c]
23A-K32	G.E./HGA		DE, DC, NO[b]	
23A-K33	G.E./HGA		DE, DC, NO[b]	
23A-K36	G.E./HGA		DE, DC, NO[b]	
23A-K37	G.E./HGA	Steam valve open signal block	DE, DC, NC	Prevents open signal to normally open valves[c]
23A-K38	G.E./HGA	Steam line break reset	DE, DC, NC	Inhibits resetting isolation[c]
23A-K43	G.E./HGA		DE, DC, NO[b]	
RCIC Relays				
13A-K3	G.E./HGA		DE, DC, NO[b]	
13A-K5	G.E./HGA		DE, DC, NO[b]	
13A-K8	G.E./HGA	Steam break reset	DE, DC, NC	Will only prevent resetting trip signal[c]

Table A.2.4-2 (continued) Initial List of Potential Low-Ruggedness Relays

Relay	Manufacturer/ Type	Function	Relay[a] Configuration	Comments
13A-K13	G.E./HGA	Low RCIC flow signal	DE, DC, NC	Would only result in closure of minimum flow valve, which is not required for system operation[c]
13A-K14	G.E./HGA		DE, DC, NO[b]	
13A-K17	G.E./HGA		DE, DC, NO[b]	
13A-K21	G.E./HGA			Spare[c]
13A-K23	G.E./HGA	Steam break reset	DE, DC, NC	Will only prevent resetting trip signal[c]
13A-K24	G.E./HGA	Testable check valve indication	DE, DC	Indication only[c]
13A-K26	G.E./HGA	Auto isolation signal	DE, DC, NC	Prevents sending open signal to normally open valve[c]
13A-K28	G.E./HGA		DE, DC, NO[b]	
13A-K29	G.E./HGA		DE, DC, NO[b]	
13A-K30	G.E./HGA		DE, DC, NO[b]	
ADS Relays				
2E-K4A/B	G.E./HGA		DE, DC, NO[b]	
2E-K8(A-D)	G.E./HGA		DE, DC, NO[b]	
2E-K10A/B	G.E./HGA		DE, DC, NO[b]	
2E-K12A/B	G.E./HGA		DE, DC, NO[b]	
RHR Relays				
10A-K23A/B	G.E./HGA	Recirc pump A Δ-pressure	DE, DC, NO[b]	
10A-K24A/B	G.E./HGA	Recirc pump A Δ-pressure	DE, DC, NO[b]	
10A-K25A/B	G.E./HGA	Recirc pump B Δ-pressure	DE, DC, NO[b]	
10A-K26A/B	G.E./HGA	Recirc pump B Δ-pressure	DE, DC, NO[b]	
10A-K27A/B	G.E./HGA	Recirc pumps running logic	DE, DC, NC	Could result in failure to select an injection path.

Table A.2.4-2 (continued) Initial List of Potential Low-Ruggedness Relays

Relay	Manufacturer/ Type	Function	Relay[a] Configuration	Comments
10A-K31A/B	G.E./HGA	Low pressure signal to break detection logic	DE, DC, NO[b]	
10A-K32A/B	G.E./HGA	Low pressure signal to break detection logic	DE, DC, NO[b]	
10A-K33A/B	G.E./HGA	Low reactor pressure logic	DE, DC, NO[b]	
10A-K35A/B	G.E./HGA	Riser Δ-pressure signal	DE, DC, NO[b]	
10A-K36A/B	G.E./HGA	Riser Δ-pressure signal	DE, DC, NO[b]	
10A-K60A/B	G.E./HGA	Drywell pressure signal to containment spray control	DE, DC, NO[b]	
10A-K63A/B	G.E./HGA	Open permissive signal to MO-2014/-2015	DE, DC	NO - Close signal to MO-2014/-2015 NC - Inhibit open signal to MO-2014/-2015
10A-K64A/B	G.E./HGA	Loop select reset	DE, DC, NC	Will not impact system operation[c]
10A-K65A/B	G.E./HGA	Low reactor pressure signal	DE, DC, NO[b]	
10A-K66A/B	G.E./HGA	Open signal to MO-2014	DE, DC, NO[b]	
10A-K67A/B	G.E./HGA	Open signal to MO-2015	DE, DC, NO[b]	
10A-K69A/B	G.E./HGA	Containment spray control	DE, DC	NO - Seal-in[b] NC - Indication light[c]
10A-K72A/B	G.E./HGA	Low pressure initiation signal	DE, DC, NO[b]	
10A-K75A/B	G.E./HGA	MO-2014 shutdown control	DE, DC	NO - Seal-in[b] NC - Indication light[c]
10A-K79A/B	G.E./HGA	Heat exchanger bypass auto control	DE, DC	NO - Heat exchanger bypass open signal[b] NC - Heat exchanger bypass close signal[c]
10A-K80A/B	G.E./HGA	CV 1994/1995 air supply manual control	DE, DC, NO[b]	
10A-K81A/B	G.E./HGA	CV 1996/1997 air supply manual control	DE, DC, NO[b]	

Table A.2.4-2 (continued) Initial List of Potential Low-Ruggedness Relays

Relay	Manufacturer/ Type	Function	Relay[a] Configuration	Comments
10A-K82A/B	G.E./HGA	Containment spray permissive alarm	DE, DC, NO[b]	
10A-K83A/B	G.E./HGA	CV 1728 position indication	DE, DC, NO[b]	
10A-K84A/B	G.E./HGA	Logic bus power monitor	EN, DC, NO[b]	
10A-K85A/B	G.E./HGA	Pump running signal to ADS	DE, DC, NO[b]	
10A-K86A/B	G.E./HGA	Drywell pressure signal to containment spray control	DE, DC, NO[b]	
10A-K88A/B	G.E./HGA	Break detection seal-in	DE, DC, NO[b]	
10A-K89A/B	G.E./HGA	Low pressure signal for break detection logic	DE, DC, NO[b]	
10A-K90A/B	G.E./HGA	Containment spray valve control	DE, DC	NO - Seal-in reset[b] NC - Indication[c]
10A-K91A/B	G.E./HGA	Low pressure open permissive for MO-2012/-2013/-2014/-2015	DE, DC, NO[b]	
10A-K92A/B	G.E./HGA	Pump running signal to ADS	DE, DC, NO[b]	

a. DE = normally deenergized, EN = normally energized, DC = powered by direct current, AC = powered by alternating current, NO = normally open, NC = normally closed

b. This relay can be eliminated from further consideration based on its actual configuration.

c. This relay can be eliminated from further consideration based on the fact that its failure would not impact successful system operation.

d. This relay can be eliminated from further consideration based on the fact that off site power is assumed unavailable.

Table A.2.4-3 **Summary - Low Ruggedness Relays**

Relay	Manufacturer /Type	Function	Relay[a] Configuration	Comments
10A-K27A	G.E./HGA	Recirc pumps running logic	DE, DC, NC	Could result in failure to select an injection path.
10A-K27B	G.E./HGA	Recirc pumps running logic	DE, DC, NC	Could result in failure to select an injection path.
10A-K63A	G.E./HGA	Open permissive signal to MO-2014/-2015	DE, DC	NO - Close signal to MO-2014/-2015 NC - Inhibit open signal to MO-2014/-2015
10A-K63B	G.E./HGA	Open permissive signal to MO-2014/-2015	DE, DC	NO - Close signal to MO-2014/-2015 NC - Inhibit open signal to MO-2014/-2015

a. DE = normally deenergized, EN = normally energized, DC = powered by direct current, AC = powered by alternating current, NO = normally open, NC = normally closed

A.2.5 Results

Those components that were not screened out by the walkdowns (Section A.2.2) or the conservative, screening evaluations (Section A.2.4) are listed in Table A.2.5-1. This screening was conducted following the EPRI NP-6041-SL guidelines for a seismic margins earthquake having a peak 5% damped spectral acceleration of 0.8g.

Each of the components in Table A.2.5-1 is dispositioned either by the SQUG effort or by evaluating the effects of the failure of the component, as described in this section. This section concludes with a summary of the systems expected to be available to provide adequate core cooling and containment pressure control following a seismic event.

A.2.5.1 Disposition of Components Needing Additional Evaluation

After Supplement 5 to Generic Letter 88-20 was issued, NSP elected to complete the seismic IPEEE for Monticello with an evaluation equivalent to a reduced scope seismic margins assessment with an additional focus on certain key components. Following the guidance of NUREG-1407 [19], outliers for reduced scope plants should be evaluated by the provisions of the Generic Implementation Procedures (GIP) [20] if the plant is also in the SQUG program. Elements outside of the SQUG program should be evaluated following requirements of the plant FSAR. The disposition of the components in Table A.2.5-1 using these reduced scope seismic margin assessment criteria is discussed here.

A.2.5.1.1 Disposition Based on SQUG Program Results

Diesel Generator 11 and 12 Air Receivers: The horizontally-oriented air receivers are mounted to their supports by U-bolts. The U-bolts were observed to be loose in the walkdown and their pre-tension, if any, is considered to be unreliable. This is to be corrected by the SQUG program, so the air receivers will be acceptable for at least the SSE after this is corrected.

4.16KV Buses 14 and 16: The conservative screening evaluation of these buses indicated a potential for buckling of their top braces. A further evaluation done under the SQUG program has shown these buses to be acceptable for the SSE.

Motor Control Centers (MCC) 42A/B, 43A/B, D311, and D312: These MCCs are supported off the floor by inverted base channels and are bolted to the floor by anchors which pass through these base channels. Their anchorage was identified as requiring further review. These MCCs were further evaluated under the SQUG program and have been found to be acceptable for the SSE.

Core Spray Pumps P-208A/B and RHR Pumps P-202A/B/C/D: These pumps are anchored by grouted in-place anchor bolts. The screening evaluation conservatively assumed that these anchor bolts were installed without procedures that would ensure significant pullout capacity. This was further evaluated under the SQUG program, and these pumps have been found to be acceptable for the SSE.

Relay Panel C32: This panel could strike an adjacent HVAC duct during an earthquake, possibly causing the relays mounted on the panel to chatter. This is to be corrected by upgrade under the SQUG program, and the panel will be acceptable for the SSE after the upgrade.

Battery Chargers D52, D53, D54, D70, D80, and D90: These battery chargers are supported on sheet metal channels. The conservative screening evaluations identified a potential for failure of the channels in their transverse direction. Further review under the SQUG program found these channels to be acceptable for the SSE.

Relay Panel C30: This panel could be subjected to earthquake-induced impact with adjacent panel C-289A. These panel have a slight separation gap. Impact could cause the relays mounted on the panel to chatter. Further review under the SQUG program showed that the separation is sufficient to prevent impact in an SSE.

Control Room Ceiling: The control room ceiling consists of acoustical tiles supported by T-bar runners. The T-bars are supported by suspension wires nailed into the overhead concrete slab. Seismic bracing is not provided. The T-bar connections lack sufficient capacity to prevent them from pulling apart, and the light fixtures are supported on the T-bars without independent safety wiring. Similar ceiling systems have collapsed in previous earthquakes. As part of the SQUG program, the lights are to be independently supported from the floor above. Though the T-bars and acoustical tile could fall, they are unlikely to cause injury to the operators or functional impairment of the control panels. The control room ceiling will be acceptable for the SSE after the upgrade.

RHR Room B HVAC Unit Cooler: The room cooler has relatively weak, vibration-isolated supports. The cooler itself is not essential to plant safety, but if its supports failed it could fall and strike the adjacent core spray pump P-208B and RHR pumps P-202B/D. Further analysis under the SQUG program showed that these supports are in fact acceptable for an SSE earthquake, largely because this equipment is located below grade and is therefore subjected to less severe seismic loads.

A.2.5.1.2 Disposition Based on Systems Analysis

Air Start Air Compressors for Diesel Generator 11 and 12: During the walkdown, the batteries for these air compressors were observed to be unanchored, allowing them to slide or overturn due to earthquake motion. A review of the system showed that the failure of these batteries to be of no concern in an earthquake. The air start motors for the diesel generators receive their pneumatic supply from air receivers which normally are pressurized; the air compressors are used to maintain the pressure in the air receivers. If the air start motors failed in an earthquake, the diesel generators would still start, since they rely only on the air receivers, which are already pressurized. Moreover, the air compressors do not normally rely on these batteries in order to start; the batteries merely supply starting power for a small backup diesel which can be manually aligned to operate the air start compressors if necessary. The failure of these batteries during an earthquake is therefore not a concern.

Condensate Storage Tanks CST-11 and CST-12: The condensate storage tanks are flat-bottomed vertical tanks enclosed within a concrete basin. Their anchorage consists of steel hold-down plates bearing on the tank bottom plate that are anchored to the concrete foundation by expansion bolts. This anchorage provides minimal resistance against tank uplift due to seismic overturning moment. The CSTs are one of two sources of water available for the operation of HPCI and RCIC. The other source is the suppression pool, which will be available to provide suction to HPCI and RCIC. The CSTs are therefore not required to be a part of the safe shutdown list for seismic events. The CRD pumps also may take suction from the CSTs, but they are not credited following a seismic event because they are load shed on loss of offsite power and an ECCS signal, so the availability of the CSTs is not relevant for them.

Control Rod Drive Pump P-201B: During the walkdown, the CUNO filter adjacent to this CRD pump was observed to be unanchored, raising a concern that this filter could overturn onto the pump recirculation line due to earthquake motion. However, the CRD system was considered to be unavailable following an earthquake regardless of whether the filter damaged the line. As noted above, the suction supply for the CRD pumps (CSTs) is not considered to be available following a seismic event. Moreover, because the CRD pumps are load shed on a loss of offsite power with a coincident ECCS signal, they are assumed to be disabled due to a loss of offsite power caused by the earthquake. Because several other means of supplying water to the reactor are expected to be available following an earthquake, such as HPCI, LPCI, core spray, and RHRSW, the CRD pumps need not be included on the seismic equipment list.

Diesel Fire Pump P-105: The batteries for this pump are not anchored and could slide or overturn due to earthquake motion, preventing the pump from starting. Though the diesel fire pump can be used to supply water to the reactor or the drywell and wetwell sprays, this is only a backup to other, normal means of providing these functions, such as with HPCI, RHR, and core spray. The fire system therefore is not required for safe shutdown following an earthquake.

Diesel Fire Pump Fuel Oil Supply Tank T-100: The tank is supported on structural steel framing. The tank supports are covered by fire-proofing, so it was not possible to verify that positive attachment to the framing exists. However, as discussed in the preceding paragraph, the fire system is not required for safe shutdown following an earthquake.

Reactor Building Closed Cooling Water (RBCCW) Pumps P-6A/B: During the walkdown it was noted that there are long, unsupported lengths of piping attached to these RBCCW pumps; the undamped lateral movement of these pipes during an earthquake could place significant loads on the RBCCW pumps at their points of attachment. This was identified as a possible means of failing the pump anchorage. The consequences of RBCCW failure following an earthquake therefore were examined. The RBCCW system provides cooling to the containment drywell coolers and the CRD pump coolers. Because the containment pressure and temperature can be controlled using RHR, RHRSW, or the containment vents, the drywell coolers are not needed for safe shutdown following a seismic event. Also, as discussed above, since the HPCI, RHR, core spray, and RHRSW systems can supply water to the reactor following an earthquake, the CRD system is not needed to supply makeup, and in any case is expected to be unavailable for other reasons. Therefore, since neither the CRD pumps nor the drywell coolers are needed following an earthquake, RBCCW also is not needed for safe shutdown following a seismic event.

Reactor Building Closed Cooling Water (RBCCW) Heat Exchangers E-5A/B/C: Seismic forces in the longitudinal direction are resisted by only one of the two saddle supports, and the conservative screening evaluation identified this as a potential means of shear failure to the anchor bolts at this support. However, as discussed in the preceding paragraph, the RBCCW system is not needed for safe shutdown following an earthquake.

Service Water Automatic Strainer F-101: Anchor bolts pass through the holes in the supports for this strainer, but the nuts which should attach the bolts to the strainer supports were not installed. The strainer could lift up due to seismic overturning moment and thereby fail the attached piping. The service water system provides cooling to components in a number of systems credited in the internal events PRA, such as the feedwater pumps, instrument air compressors, and RBCCW heat exchangers. However, none of these systems is credited in the seismic IPEEE analysis. The feedwater system is not expected to be available following an earthquake due to the assumed loss of offsite power. Instrument air conservatively was not included in the IPEEE equipment list to avoid having to walk down the system, which has piping and components distributed throughout the plant. RBCCW operation is not considered to be important following an earthquake for the reasons noted earlier. Service water is also a backup to the EDG ESW system and the emergency service water (ESW) system. However, since all the components in both the EDG and EDG ESW systems were found to be seismically rugged, this backup function of the service water system is not important following an earthquake. The service water system is therefore not needed for safe shutdown of the plant following a seismic event.

A.2.5.2 Safe Shutdown Functions Following a Seismic Event

All the functions that are needed to ensure adequate core cooling and containment pressure control following an earthquake have multiple and, in some cases, diverse trains of equipment, each one of which is capable of performing that function. Each train of equipment has been shown to have a high confidence of being operable following seismic events at the SSE level. As a result, it is concluded that the Monticello plant has no vulnerability to seismic events.

Reactivity Control

All systems, structures, and components (SSCs) that provide reactivity control were found to be seismically rugged and would be available following an earthquake.

Given the loss of offsite power or small LOCA that is assumed to follow an earthquake, several independent signals would be expected to induce a reactor trip, such as high reactor pressure, high flux, turbine stop valve closure, or high containment pressure The principal means of reactor shutdown would be the reactor protection system and control rod insertion.

Though it is very unlikely that during an earthquake there would be a coincident failure to scram, the backup systems that would then be used to shut down the reactor would also be available following an earthquake. All of the components associated with the auxiliary rod insertion system, recirculation pump trip, and standby liquid control system were found to be seismically rugged.

High Pressure Injection

All the SSCs associated with high pressure injection were found to be seismically rugged or are being upgraded to assure a low potential of failure at the SSE.

Both HPCI and RCIC would be available to provide makeup at high reactor pressure during the loss of offsite power assumed to follow an earthquake. The suction of these systems would be aligned to the suppression pool if the CSTs failed as a result of an earthquake. If a small LOCA also occurred, HPCI would be the principal source of high pressure makeup due to its higher capacity (approximately 3000 gpm as compared to 400 gpm for RCIC).

Reactor Pressure Control

All the SSCs associated with reactor pressure control were found to be seismically rugged.

Should high pressure injection systems be unavailable, the reactor vessel would be depressurized to allow the low pressure systems to inject. This is done by opening at least one of the eight SRVs.

Low Pressure Injection

All the SSCs associated with low pressure injection were found to be seismically rugged or were shown to be acceptable at the SSE.

A number of redundant systems are available following a seismic event to provide low pressure makeup to the reactor, including RHR in the low pressure coolant injection mode and core spray. Each of these systems has two trains, with any one train capable of providing adequate core cooling.

In addition to the RHR and core spray systems, the RHRSW system could also be used to supply water to the reactor. This means of injection would require local manual crosstie of the RHRSW pumps to Loop A of RHR and would be initiated only if LPCI and core spray were not effective.

Containment Pressure Control

All the SSCs associated with containment pressure control were found to be seismically rugged or were shown to be acceptable at the SSE.

Several redundant systems are available following a seismic event to control pressure in the containment. The preferred source of containment heat removal is the RHR system in the suppression pool cooling or shutdown cooling modes. Either one of the two trains of RHR is capable of removing decay heat from the containment. If RHR were not available, the drywell sprays or wetwell sprays could be initiated using RHRSW as an injection source. If all of these modes of containment pressure control were unavailable, the containment could still be vented using the hard-piped vent. This would only be initiated to prevent the containment from exceeding its design pressure of 56 psig, which would not be reached for approximately a day, assuming no other means of containment pressure control was available.

AC Power

All SSCs associated with AC power either were found to be seismically rugged, were shown to be acceptable at the SSE, or are to be upgraded to the SSE level.

Both divisions of essential AC power would be available following a seismic event. Each division would be supplied by a diesel generator if a loss of offsite power followed the seismic event.

DC Power

All SSCs associated with DC power were found to be seismically rugged or were shown to be acceptable at the SSE.

Both divisions of 250V DC and 125V DC power would be available following a seismic event. The 250V DC divisions I and II support RCIC and HPCI, respectively. Division II 250V DC also supplies power to the hard-piped vent solenoid through an inverter. 125V DC power divisions I and II supply power for their respective EDG circuitry, AC breaker control, and RCIC and HPCI instrument and control. 125V DC power is also necessary for operation of four of the eight SRVs.

Nitrogen

All SSCs associated with the bottled nitrogen supply to the SRVs were found to be seismically rugged.

Bottled nitrogen is the safety-related source of pneumatic pressure for operation of six of the eight SRVs for reactor depressurization.

EDG Emergency Service Water

All SSCs associated with the EDG ESW system were found to be seismically rugged.

Two trains of EDG ESW are provided for cooling the emergency diesel generators. One pump is provided for each generator, although the discharge of each pump can be aligned to supply either generator.

Emergency Service Water

All SSCs associated with the emergency filtration train ESW system were found to be seismically rugged.

The purpose of the ESW system is to provide long term cooling to the ECCS pump motor coolers. There are two ESW pumps, each supplying cooling to its respective division of RHR and core spray pumps. ESW also supplies ECCS room cooling. The internal events PRA shows that this function is not necessary to support operation of the RHR and core spray pumps.

RHR Service Water

All SSCs associated with the RHRSW system were found to be seismically rugged.

The principal purpose of the RHRSW system is to provide a heat sink for the RHR heat exchangers when they are being used to cool the reactor coolant or the suppression pool. Four pumps are available, two for each RHR heat exchanger. Any one pump is sufficient to provide adequate decay heat removal, provided its associated RHR system is functioning.

A secondary purpose for RHRSW is to provide low pressure makeup to the reactor or a source of water for the wetwell and drywell sprays. This means of vessel makeup and containment pressure control would be initiated only if normal means of performing this function were not available.

As shown, all the functions that are needed to ensure adequate core cooling and containment pressure control following an earthquake have multiple and, in some cases, diverse trains of equipment, each one of which is capable of performing that function. Each train of equipment has been shown to have a high confidence of being operable following seismic events at the SSE level. As a result, it is concluded that the Monticello plant has no vulnerability to seismic events.

In addition to showing that the Monticello plant could be brought to safe shutdown through various means following an SSE earthquake, this seismic evaluation also shows that safe shutdown can be reached even for earthquakes beyond the SSE level. The original screening of plant equipment identified those SSCs which meet the screening criteria of EPRI NP-6041-SL (equivalent to a 0.3g earthquake) and are therefore considered to be seismically rugged. As shown in Table A.2.5-2, all the safety functions needed to reach safe shutdown can be achieved using only the equipment that meets these screening criteria, with the single additional assumption that the loose U-bolts on the EDG air receiver tanks will be corrected as planned under the SQUG program (Section A.2.5.1.1).

Table A.2.5-1 Disposition of Components Not Meeting EPRI NP-6041-SL Screening Criteria

COMPONENT	SYSTEM	POTENTIAL FAILURE MODE	CONCLUSION
DG 11 and 12 Air Receivers	AC Power	Sliding-induced pipe failure. Pre-tension of U-bolts not reliable.	To be corrected by SQUG program.
DG 11 and 12 Air Start Air Compressors	AC Power	Sliding or overturning of unanchored batteries.	Not required to start diesel generators.
4.16KV Buses 14 and 16	AC Power	Buckling of top brace.	Acceptable at SSE per SQUG program.
MCC 42A/B, 43A/B	AC Power	Anchorage. Anchor bolts pass through inverted base channel.	Acceptable at SSE per SQUG program.
Condensate Storage Tanks CST-11 and 12	Condensate Storage	Minimal anchorage.	Suppression pool provides adequate source of makeup for ECCS pumps.
CRD Pump P-201B	Control Rod Drive	Overturning of adjacent, unanchored CUNO filter onto pump recirculation line.	Not credited; redundant to other high and low pressure injection systems included in IPEEE scope.
Core Spray Pumps P-208A/B	Core Spray	Grouted in-place anchor bolts.	Acceptable at SSE per SQUG program.
Relay Panel C32	Core Spray	Relay chatter due to impact with HVAC duct behind panel.	To be corrected by SQUG program.
Battery Chargers D52, D53, D54	DC Power	Transverse bending of sheet metal support channels.	Acceptable at SSE per SQUG program.

Table A.2.5-1 (continued) **Disposition of Components Not Meeting EPRI NP-6041-SL Screening Criteria**

COMPONENT	SYSTEM	POTENTIAL FAILURE MODE	CONCLUSION
Battery Chargers D70, D80, D90	DC Power	Transverse bending of sheet metal support channels.	Acceptable at SSE per SQUG program.
MCC D311	DC Power	Anchorage. Anchor bolts pass through inverted base channel.	Acceptable at SSE per SQUG program.
Diesel Fire Pump P-105	Fire Protection	Sliding or overturning of unanchored batteries.	Not credited; redundant to other high and low pressure injection systems included in IPEEE scope.
Diesel Fire Pump Fuel Oil Supply Tank T-100	Fire Protection	Sliding or overturning of tank causing failure of attached piping. Attachment of tank to support obscured by insulation, assumed no weld.	Not credited; redundant to other high and low pressure injection systems included in IPEEE scope.
MCC D312	High Pressure Coolant Injection	Anchorage. Anchor bolts pass through inverted base channel.	Acceptable at SSE per SQUG program.
RBCCW Pumps P-6A/B	Reactor Building Closed Cooling Water	Significant nozzle loads from long, unsupported lengths of attached piping may fail anchor bolts.	Not credited; other means of providing contanment control are included in IPEEE scope.
RBCCW Heat Exchangers E-5A/B/C	Reactor Building Closed Cooling Water	Anchor bolts	Not credited; other means of providing containment control are included in IPEEE scope.
Relay Panel C30	Reactor Core Isolation Cooling	Relay chatter due to impact with adjacent panel C-289-A (small separation gap).	Acceptable at SSE per SQUG program.

Table A.2.5-1 (continued) Disposition of Components Not Meeting EPRI NP-6041-SL Screening Criteria

COMPONENT	SYSTEM	POTENTIAL FAILURE MODE	CONCLUSION
RHR Pumps P-202A, B, C, D	Residual Heat Removal	Grouted in-place anchor bolts.	Acceptable at SSE per SQUG program.
Service Water Automatic Strainer F-101	Service Water	Anchor bolts lack nuts to hold down strainer.	Not credited; redundant to other systems included in IPEEE scope.
Control Room Ceiling	-	Ceiling collapse. Ceiling system unbraced, vulnerable T-bar connections, light fixtures not safety-wired.	Light fixtures to be upgraded under SQUG program. Falling T-bars and acoustical tiles unlikely to cause damage or injury.
RHR Room B HVAC Unit Cooler	-	Failure of weak, vibration-isolated supports. Cooler is non-essential, but could fail and impact Pumps P-202B/D and 208B.	Acceptable at SSE per SQUG program.

Table A.2.5-2 **Monticello Safety Function Performance Beyond the SSE**

Function	Comment
Reactivity Control	RPS and control rod insertion.
High Pressure Injection	HPCI and RCIC not credited because MCC 311 and 312 do not meet screening criteria.
Reactor Depressurization	All eight SRVs available short-term. SRVs A - D available long term (after 4 hours); SRVs E - H depend on battery chargers D52, 53, 54, 70, 80 and 90 which do not meet screening criteria.
Low Pressure Injection	RHR and core spray pumps do not meet screening criteria and are not credited. RHRSW pumps A and C are available.
Containment Pressure Control	RHR pumps do not meet screening criteria, so suppression pool cooling and shutdown cooling are not credited. The hard-piped vent depends on Division II power and therefore is not credited. RHRSW pumps A and C are available to provide wetwell and drywell spray.
Support Systems	
AC Power	Division I power available. Division II power is not available; it depends on Bus 16 which does not meet the screening criteria.
DC Power	Both divisions of 250V and 125V DC are available short term (4 hours). Division I 250V and 125V DC available long term.
Nitrogen	Bottled nitrogen is available for SRV operation.
EDG ESW	Division I EDG ESW is available.
Emergency Service Water	Division I ESW is available.
RHR Service Water	Division I RHR service water is available.

A.2.6 Analysis of Containment Performance

As indicated in NUREG-1407, the focus of the containment evaluation is to identify any severe accident issues unique to seismic events that may involve early failure of important containment functions. The containment evaluation for Monticello revealed no such issues. The purpose of this section is to review and discuss the containment response following a seismic event and to present the details of the containment-related evaluations that were performed for the seismic IPEEE.

A.2.6.1 Basis for the Scope of the Analysis

The scope of this containment analysis is based upon a review of the Level 2 analysis in the internal events PRA, as well as the specific issues presented in Section 3.2.6 of NUREG-1407. The focus of the evaluation was to identify any potential early containment failure modes unique to seismic events that had not already been evaluated as a part of the internal events PRA.

The NUREG-1407 guidance requires an evaluation of any seismically induced containment failures and other containment performance insights. Particularly, it should consider vulnerabilities found in the systems and functions which could lead to early containment failure or which may result in high consequences. These include containment isolation, bypass, and integrity, and systems required to prevent early failure.

A.2.6.2 Containment Structures and Systems

A seismic assessment was performed to identify any vulnerability that could lead to early failure of containment functions. The structures, systems, and components needed to ensure containment integrity, containment isolation, and prevention of bypass were reviewed.

Containment Structures

The containment structures and components were evaluated as described in Section A.2.5.1. The drywell was found to satisfy the EPRI NP-6041-SL screening criteria. A review of the Plant Unique Analysis Report [29] found that the stresses acting on the containment components due to seismic loads are low compared to those from other loads, with the exception of the suppression chamber seismic restraints. A seismic evaluation of these restraints determined that they have adequate seismic capacity.

A general inspection of the containment penetrations and other components inside containment did not identify any significant seismic vulnerabilities. The equipment hatch, personnel airlock, and containment penetrations do not have inflatable seals or cooling systems.

Containment Systems

Systems important to maintaining containment integrity after a core damage event were identified in the Monticello internal events PRA. A summary of these systems and the functions that they provide follows:

Containment Isolation
- Isolation Valves

Combustible Gas Control
- Inerting (Primary containment atmosphere control system)

Debris Cooling
- RHR (LPCI mode)
- Core spray
- RHRSW

Containment Pressure Control
- RHR (suppression pool cooling mode)
- Hard-piped vent
- RHRSW (drywell and wetwell sprays)

Radioactive Release Control
- Hard-piped vent (fission product scrubbing in the suppression pool)

For components in many of these systems, a screening evaluation was done as part of the plant walkdowns and seismic margins assessment, as discussed in Sections A.2.2, A.2.4, and A.2.5. The functions and systems listed above were reviewed to determine whether all systems which are important to containment performance were evaluated during the seismic margins assessment.

In this containment evaluation, any system or component which must be disabled in order to reach core damage was not credited as a means of avoiding containment failure. Table A.2.6-1 summarizes the systems which would be available to provide functions such as debris cooling and containment heat removal.

The accident sequence types defined in the internal events PRA are presented below. Each discussion supports the conclusions that (1) the majority of systems important to containment performance under severe accident conditions were considered as a part of the seismic margins assessment, and (2) the containment response to core damage following a seismic event is similar to that analyzed in the internal events PRA.

Containment Response

Seven accident classes or accident sequence types were defined in the internal events PRA. These include:

Class 1A	Transients in which core damage occurs at high reactor pressure
Class 1D	Transients in which core damage occurs at low reactor pressure
Class 1B	Station blackout events
Class 3B	LOCAs in which core damage occurs at high reactor pressure
Class 3C	LOCAs in which core damage occurs at low reactor pressure
Class 2	Events in which core damage occurs because containment pressure control fails
Class 4	Events in which core damage because of a failure to scram (ATWS)

The two initiating events which may accompany a seismic event are a loss of offsite power or a small loss of coolant accident. These initiators are associated with the first five of the accident classes given above. In these accident classes, core damage would occur while the containment is still intact. For each of these five accident classes, a comparison between the plant response as analyzed in the internal events PRA and the response that would be expected if the accident were initiated by an earthquake is described below.

Transient or LOCA at High Reactor Pressure (Classes 1A and 3B)

For both of these accident classes, core damage is assumed to occur as a result of the loss of all high pressure injection systems coincident with a failure to depressurize the reactor through the SRVs. If no high pressure injection system is recovered before the core melts through the vessel lower head, then the reactor would depressurize when the lower head is breached. Low pressure coolant makeup systems would then be able to cool the debris on the drywell floor. Many of the same systems that were considered in the internal events PRA are available for debris cooling following a seismic event. These low pressure systems include RHR in the LPCI mode, core spray, and RHRSW. The same systems credited in the internal events PRA for long-term decay heat removal would also be available following an earthquake: RHR in the suppression pool cooling mode and the hard-piped vent. Due to the number of systems available to provide debris cooling and decay heat removal, only a limited fraction of the sequences initiated by transients or LOCAs and having high reactor pressure would be expected to lead to either early or late containment failure. The containment response to core damage events in these accident classes is expected to be the same regardless of whether the accident is initiated by an earthquake.

Transient or LOCA at Low Reactor Pressure (Classes 1D and 3C)

For these accident classes, core damage is assumed to occur as a result of the loss of all high pressure and low pressure injection systems. The reactor is depressurized through the break if the initiator is a LOCA or through the SRVs if it is not. If no injection system is recovered, the damaged core would eventually penetrate the vessel lower head and fall into the drywell, where it would remain uncooled because no injection system is available. As discussed in the internal events PRA, most of the debris would remain in the large sumps located directly beneath the reactor vessel, so there is little potential for liner meltthrough to cause an early containment failure. However, the containment pressure and temperature would rise over time as core concrete interaction occurred. The internal events PRA shows that under these conditions the containment design pressure would not be reached for at least 12 hours, and the containment failure pressure would not be reached for 24 hours. If no means of cooling the debris or removing heat from the containment could be recovered before the design pressure were reached, the hard-piped vent would be used to relieve containment pressure, remove decay heat from the containment, and control radioactive releases by scrubbing in the suppression pool. There is little potential for early containment failure. It is concluded that the plant and containment responses to these accident classes when initiated by an earthquake is similar to that described in the internal events PRA.

Station Blackout

The potential for core damage due to station blackout sequences results primarily from battery depletion. The containment would be intact at the time of core damage for this type of accident. If no AC power source is recovered within the first six to eight hours of the blackout, the uncooled core debris in the vessel could melt through the lower head and enter the drywell. Most of the core debris would remain in the large containment sumps directly beneath the vessel, so there is little potential for early containment failure due to liner melt-through. If no means of cooling the debris or removing decay heat from the containment is recovered, the containment would eventually pressurize; however, this would take place so slowly that it would take roughly a day or more to reach the containment failure pressure. Unless an AC power source is recovered, releases could not be controlled by venting because battery depletion would prevent operation of the valves in the vent line, which depend on DC power. The timing and response of the containment to severe accident conditions associated with a seismically initiated station blackout are very similar to the internal events PRA.

The remaining two accident sequence types, containment pressure control failure and ATWS, are assumed to lead to containment failure at the time of core damage. As a result, the challenge to containment for these accident sequence types has been reviewed as a part of the evaluation of the functions important to providing adequate core cooling.

Containment Pressure Control Failure

Failure of containment pressure control does not result in early containment failure. The challenge to containment for sequences in this accident class is from the gradual pressurization of containment due to decay heat generation. A complete loss of decay heat removal was analyzed for the internal events PRA and indicates that more than a day is required to reach the containment failure pressure as a result of decay heat. This relatively long time frame applies to seismically induced initiators as well.

Anticipated Transient Without Scram (ATWS)

An ATWS would cause an early challenge to the containment as a result of steam pressurization following a failure to scram. The timing and consequences of this type of challenge are no different if the event is seismically initiated. Furthermore, the potential for an ATWS to occur at the same time as a seismic event is extremely small.

Containment Isolation

Isolation valves are provided on al lines penetrating the drywell and suppression chamber to assure the integrity of the containment under accident conditions. Those isolation valves which must be closed to assure containment integrity immediately after a major accident are automatically controlled by the plant protection system.

Many different types of penetrations were considered during the containment isolation evaluation of the internal events PRA. The following piping and hatch penetration groups were examined:

1. Feedwater, main steam lines, and associated main steam drain lines
2. HPCI steam lines
3. RCIC steam lines
4. CRD lines
5. Low pressure ECCS lines (for interfacing systems LOCA considerations)
6. Instrument lines
7. Personnel locks, hatches, and the drywell dome
8. Cable penetrations
9. Instrument air lines
10. RBCCW lines
11. Purge and vent lines

The penetrations listed in items 1 through 6 above are important primarily when analyzing the potential for containment bypass or interfacing systems LOCAs, because breaks or leaks in such lines could result in releases directly from the reactor vessel into the plant buildings. However, since these piping systems are seismically rugged they do not contribute significantly to the potential for containment bypass following a seismic event.

The penetrations and piping in groups 7 through 11 above must be isolated to prevent flow of the containment atmosphere into the reactor building or outdoors. If radionuclides are released into the containment or the containment becomes pressurized as a result of an accident, isolating the containment minimizes any releases to the outside atmosphere and avoids potential adverse impacts on accident mitigating systems in the reactor building. The following considerations were used to help focus the review of penetrations and piping in these groups:

- Penetrations of closed piping systems: A system which is not open to the containment atmosphere - a "closed" system - cannot provide a pathway from the containment atmosphere to the outside unless (1) its isolation valves fail, and (2) the piping itself breaks so that the containment atmosphere can enter the system piping and be released through the open isolation valves. Because all of the components in such systems were found to be seismically rugged, neither of these failures is expected to occur as a result of an earthquake. Both of these failures would therefore have to occur independently and simultaneously in order to compromise the containment; the probability of this is negligibly small. For this reason, closed systems such as RBCCW are not a threat to containment integrity during an earthquake.

- Hatches, personnel locks, drywell dome: These items are considered to be part of the continuous liner of the containment and therefore are factored into the evaluation performed for the overall containment structural response.

- Pipes with diameters less than two inches: These pipes, such as instrument and sample lines, are not considered important for two reasons: (a) aerosol plugging following a severe accident is likely to reduce the amount of leakage which could occur through these pipes, and (b) they are not large enough to relieve containment pressure fast enough to prevent eventual failure of the containment by overpressurization.

Table A.2.6-2 shows the containment penetrations which were not screened out using the criteria given above. The table shows the configuration of these containment isolation valves, their normal positions, signals which close the valves, and dependencies of the valves on support systems for motive and control power.

The isolation valves in this table are the same as those considered in the interval events PRA. It should be noted that all of these isolation valves are either normally closed or they fail closed on the loss of air or power. These valves are designed such that the potential for containment isolation following a seismic event is high and can be considered similar to that evaluated for the internal events PRA.

Table A.2.6-1 Monticello Level 1 to Level 2 Dependencies

	Debris Cooling									Reactor Depress.	Containment Pressure Control		
Accident Class	Feed-water[1]	HPCI	RCIC	CRD[2]	Conden-sate[1]	LPCI	CS	Fire System	Cont. Spray[3]	SRVs	Main Cond.[1]	RHR	Vent
1A		--	--			✓	✓	✓		--		✓	✓
1B[4]		--	--			--	--	✓		--		--	--
1D		--	--			--	--	✓[5]		N/A		✓[6]	✓
2		✓	--[7]			✓	✓	✓		--[8]		N/A	N/A
3B		--	✓[9]			✓	✓	✓		--		✓	✓
3C		--	--			--	--	✓[5]		N/A		✓[6]	✓
4		--[10]	--[10]			--[10]	--[10]	✓[11]		✓		N/A	N/A

✓ Credited in Level 2.
-- Failed as part of Level 1.
N/A Not relevant to outcome of sequence.

1 Not credited following a loss of off-site power.

2 Load shed on loss of off-site power and ECCS signal.

3 Use prohibited by Drywell Spray Initiation Limit curve.

4 Off-site and on-site AC power recovery is not credited.

5 Potentially available, given additional time to align.

6 RHR highly dependent on LPCI.

7 Failed due to high turbine exhaust pressure.

8 Assumed to be failed due to containment pressure greater than 70 psig.

9 Available unless core debris penetrates lower vessel head.

10 Assumed to fail due to environment in reactor building.

11 Sufficient for debris cooling once subcritical.

Table A.2.6-2 Contributors to Containment Isolation Failure

DESCRIPTION	PENETRATION NUMBER	SIZE	CONFIGURATION	POSITION	SIGNALS	POWER/AIR
Torus-Reactor building vacuum breakers	X-218	20"	2 parallel paths of 1 AOV and 1 check valve in series	normally closed	Open on containment pressure of -10" H_2O	AOVs fail open on loss of air or AC
Torus ventilation supply	X-218	18"	2 AOVs in series	normally closed	Group 2	AOVs fail closed on loss of air or AC
Post-LOCA recombiner return	X-218	6"	2 AOVs in series	normally closed	Group 2	AOVs fail closed on loss of air or AC
Torus ventilation exhaust	X-205	20"	1 AOV and 1 (2") AOV in parallel, with 1 AOV in series	normally closed	Group 2	AOVs fail closed on loss of air or AC
Post-LOCA recombiner return	X-205	6"	2 AOVs in series	normally closed	Group 2	AOVs fail closed on loss of air or AC
Drywell ventilation exhaust	X-25	18"	1 AOV and 1 (2") AOV in parallel, with 1 AOV in series	normally closed	Group 2	AOVs fail closed on loss of air or AC
Drywell ventilation supply	X-26	18"	2 AOVs in series	normally closed	Group 2	AOVs fail closed on loss of air or AC
Floor sump	X-18	2"	2 AOVs in series	normally open	Group 2	AOVs fail closed on loss of air or AC
Equipment sump	X-19	2"	2 AOVs in series	normally open	Group 2	AOVs fail closed on loss of air or AC
CRD vent/drain lines (4 lines)		2"	2 AOVs in series on each of 4 lines	normally open	Group 2	AOVs fail closed on loss of air or AC

A.2.7 Conclusions and Recommendations

The significant conclusions from the seismic evaluation of essential Monticello structures and components are that:

- The Class I structures have relatively high seismic capacities. Building separations were determined to be sufficient to prevent impact between Class I structures. Damage to Class I structures due to failure of adjoining Class II structures was judged to be unlikely.
- Concrete block walls either (1) were found to have sufficient seismic capacity or (2) are unlikely to damage essential components even if they fail.
- Mechanical equipment, electrical equipment, heat exchangers, and certain tanks are adequate to retain their structural integrity and post-earthquake functionality. Those tanks that were not found to be adequate were shown not to have significant consequences if they fail.
- Distributed systems such as piping, cable trays, conduit, and HVAC ducting were verified to be seismically adequate.
- Most "bad actor" relays that were within the scope of the seismic IPEEE either (1) were shown to have configurations which make them unsusceptible to chatter or (2) the consequences if the relay failed were not significant. All relays for which corrective action may be necessary are within the scope of the SQUG program and will be addressed as part of that program.

The majority of the components included in the seismic margins assessment for Monticello are seismically rugged and meet the screening criteria in EPRI NP-6041-SL. Most of the components that fell below the EPRI screening criteria are within the scope of the SQUG program and either were shown to be adequate at the SSE or are to be corrected under the SQUG program. The remaining components which are outside the SQUG program and do not meet the EPRI screening criteria were shown to be of limited importance in preventing or mitigating accidents which could be caused by a seismic event.

In addition, it was shown that even for earthquakes beyond the SSE level, the Monticello plant could be brought to safe shutdown using only the equipment which meets the EPRI screening criteria to achieve reactor trip, reactor depressurization, a source of low pressure injection, and long term containment pressure control.

The containment response to a severe accident following a seismic event is expected to be similar to that analyzed for the internal events PRA. No early containment failure modes unique to a seismic event were identified as a part of this analysis.

A.2.8 Unresolved Safety Issues and Other Seismic Safety Issues

1. USI A-17 [21] systems interactions were considered in the IPEEE seismic walkdowns and seismic margin evaluations. Any significant seismic systems interactions were identified in the IPEEE walkdowns. USI A-17 is concerned with operational dependencies between systems. A qualitative analysis of these dependencies was considered in assessing the availability of plant functions following a seismic event. Insights from the internal events PRA were instrumental in this task.

2. USI A-40 [22] includes the seismic analysis of above-ground tanks. Most tanks important to the operation of the systems credited in the seismic margins assessment were found to have sufficient seismic capacity to meet the screening criteria of EPRI NP-6041-SL. The condensate storage tanks were not credited as a water source in the seismic margins assessment and therefore were not evaluated in the IPEEE.

3. The seismic walkdowns for the IPEEE were initiated prior to walkdowns for USI A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants." The walkdown data obtained for the IPEEE was made available for USI A-46 activities.

4. USI A-45 [23] addresses the adequacy of the heat removal function at operating plants. There are four possible methods of decay heat removal at Monticello: the main condenser, containment venting, the reactor water cleanup (RWCU) system, and the RHR system in either the suppression pool cooling, wetwell spray, drywell spray, or shutdown cooling modes.

   The main condenser is the preferred decay heat removal system during a normal shutdown until the reactor pressure drops to the point where RHR shutdown cooling can be placed in service. Important support systems for the main condenser include offsite power, circulating water, condensate, instrument air, and service water. This main condenser is therefore assumed to be unavailable for decay heat removal following a seismic event.

   If the main condenser is not available, RHR suppression pool cooling would be used as an indirect decay heat removal system, removing heat from the reactor vessel via the SRVs and the suppression pool. Most components of the RHR system were found to have high seismic capacity and met the EPRI NP-6041-SL screening criteria or were shown to be adequate at the SSE. Any secondary seismic failures that could affect the operation of RHR components were identified. Suppression pool cooling was the principal mode of RHR containment heat removal credited in this analysis. Other operating modes of RHR which can remove decay heat include shutdown cooling, wetwell sprays and drywell sprays. Shutdown cooling can remove decay heat once the reactor pressure has been lowered. Wetwell or drywell sprays are initiated per the emergency operating procedures at high containment pressures and temperatures. Modes of RHR other than suppression pool cooling have a significant effect only if the suppression pool cooling valves failed. Commonalities with the remaining portion of the RHR system reduce the impact of these other modes of RHR. RHRSW can supply water to the drywell and wetwell sprays independently of many of the components in the RHR system. All modes of RHR heat removal depend on RHRSW operation.

The suppression pool temperature would rise and containment pressure would gradually increase if the main condenser and the suppression pool cooling and shutdown cooling modes of RHR were all unavailable. About a day would be required before the containment would reach its design pressure, assuming that makeup to the reactor was from the suppression pool. By that time, recovery actions would be underway to correct the failures in the RHR system. The containment sprays could be used to control pressure in the containment. If containment sprays and recovery actions were all unsuccessful, the containment would be vented at its 56 psig design pressure. The vent and purge lines are not considered to be available following a seismic event as they require instrument air for operation; however, the hard-piped vent would be available as it depends only on DC power and nitrogen for operation.

If all means of decay heat removal failed, including venting, the containment pressure would continue to increase at a slow rate driven by the decay heat rate. Two to three days are necessary to pressurize the containment to its ultimate capacity of about 103 psig. At this point, the containment is assumed to fail at the drywell head enclosure or the torus expansion bellows. Releases from these locations would primarily affect the refuel floor and the torus area. Because of these failure locations, injection systems in the turbine building and some systems in the corner rooms of the reactor building are likely to remain operable after containment failure. It is highly probable that continued injection to the vessel after containment failure will prevent core damage. (Refer to the equipment survivability discussion in Sections 4.1.3 and 4.2.2-2 of the IPE.)

Actions that could significantly prolong the event, but not necessarily prevent it, also were not credited. Those actions included use of the RWCU system, either in feed and bleed or heat removal modes.

The decay heat removal (DHR) issue was examined as part of the IPE, the details of which are contained in Sections 4.4 and 6.6 of the IPE submittal [24]. The results of this examination indicate that the loss of DHR capability does not contribute significantly to core damage. The results of seismic analysis of loss of DHR did not differ significantly from the loss of DHR evaluated in the IPE.

Monticello's means of dealing with decay heat removal during accidents involving seismic events is similar to that described in the internal events IPE and is considered adequate to resolve this generic issue.

5. Charleston Earthquake Issue:

The NRC states in Generic Letter 88-20, supplement 4, that the Charleston Earthquake Issue is subsumed in the IPEEE. NSP has performed a seismic margins assessment for the Monticello IPEEE and therefore has fulfilled the requirements for this issue.

A.2.9 References

1. U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 4, June 1991.

2. U.S. Nuclear Regulatory Commission, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Generic Letter 88-20, Supplement 5, September 8, 1995.

3. Jack R. Benjamin and Associates, et al, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)," prepared for the Electric Power Research Institute, EPRI NP-6041-SL, August 1991.

4. U.S. Nuclear Regulatory Commission, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," Generic Letter 87-02, February 19, 1987.

5. U.S. Nuclear Regulatory Commission, "Standard Review Plan," NUREG-0800, August 1989.

6. U.S. Nuclear Regulatory Commission, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60, Revision 1, December 1973.

7. Harding-Lawson Associates, "Geophysical Investigation, Monticello Nuclear Generating Station, Monticello, Minnesota," HLA Job No. 3854,058.01, September 1981.

8. Seed, H.B. and I.M. Idriss, "Soil Moduli and Damping Factors for Dynamic Response Analyses,' EERC 70-10, Earthquake Engineering Research Center, University of California, Berkeley, December 1972.

9. Schnabel, P.B., J. Lysmer, and H.B. Seed, "SHAKE - A Computer Program for Earthquake Response Analysis of Horizontally Layered Sites," EERC 72-12, Earthquake Engineering Research Center, University of California, Berkeley, December 1972.

10. Lysmer, J. et al, "SASSI - A System for Analysis of Soil-Structures Interaction," UCB/GT/81-02, University of California, Berkeley, 1981.

11. Wong, H.L. and J.E. Luco, "Soil-Structure Interaction: A Linear Continuum Mechanics Approach (CLASS I)," Report CE, Department of Civil Engineering, University of Southern California, 1980.

12. John A. Blume & Associates, Engineers, "Monticello Nuclear Generation Plant Earthquake Analysis; Reactor Building," prepared for General Electric Company, 1967a.

John A. Blume & Associates, Engineers, "Monticello Nuclear Generation Plant Earthquake Analysis; Drywell," prepared for General Electric Company, 1967b.

John A. Blume & Associates, Engineers, "Monticello Nuclear Generation Plant Earthquake Analysis; Reactor Pressure Vessel," prepared for General Electric Company, 1967c.

13. NUTECH Engineers, "Monticello Site Specific Seismic Evaluation for the Reactor and Turbine Buildings," NSP-14-101, prepared for Northern States Power Company, October 1985.

14. Newmark, N.M. and W.J. Hall, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-0098, 1977.

15. NUTECH Engineers, "Monticello Nuclear Generating Plant, Plant Unique Analysis Report," Volumes 1 to 5, NSP-74-101 to 105, prepared for Northern States Power Company, November 1982.

16. American Concrete Institute and American Society of Civil Engineers, "Building Code Requirements for Masonry Structures (ACI 530-88/ASCE 5-88) and Specifications for Masonry Structures (ACI 530.1-88/ASCE 6-88)," ACI 530-88/ASCE 5-88, 1988.

17. URS/John A. Blume & Associates, Engineers, "Seismic Safety Evaluation of the Anchorage of Safety-Related Electrical Equipment at the Monticello Nuclear Generating Plant," URS/JAB 8158, prepared for Northern States Power Company, 1983.

18. MPR Associates, "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality," prepared for Electric Power Research Institute, EPRI NP-7148-SL, December 1990.

19. U.S. Nuclear Regulatory Commission, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," NUREG-1407, May 1991.

20. Seismic Qualification Utility Group, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Power Plant Equipment," Revision 2, Corrected, February 14, 1992.

21. U.S. Nuclear Regulatory Commission, "Unresolved Safety Issue (USI) A-17, System Interactions in Nuclear Power Plants."

22. U.S. Nuclear Regulatory Commission, "Unresolved Safety Issue (USI) A-40, Seismic Design Criteria."

23. U.S. Nuclear Regulatory Commission, "Unresolved Safety Issue (USI) A-45, Shutdown Decay Heat Removal (DHR) Requirements."

24. Northern States Power Company, "Monticello Nuclear Generating Plant Individual Plant Examination (IPE)," NSPNAD-92003, Revision 0, February 1992.

Monticello
Individual Plant Examination
of External Events (IPEEE)

NSPLMI-95001

Appendix B
Revision 1

Internal Fires Analysis

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B.1 INTRODUCTION

B.1.1 Background

The assessment that is described in this appendix addresses the internal fires requirements of Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" [1], for the Monticello Nuclear Generating Plant. The fire analysis performed for the IPEEE began in 1992 and reflects plant changes made since the IPE [2]. This internal fire assessment combines the probabilistic risk assessment approach used in the IPE with the deterministic evaluation techniques of EPRI's fire induced vulnerabilities evaluation (FIVE) methodology [4].

B.1.2 Plant Familiarization

The Monticello nuclear generating plant is a low power-density BWR-3 with Mark I containment, designed by General Electric Company and built by Bechtel Corporation. The reactor core produces 1670 MWt with an electrical output of 545 MWe. The plant is located in Monticello, Minnesota. Construction started on June 19, 1967 and full commercial operation began on June 30, 1971.

Implementation of the requirements of 10 CFR 50, Appendix R, contributed significantly to the low overall risk due to fires at the Monticello plant. These requirements addressed issues such as fire barriers and penetration seals, administrative control of combustibles, fire brigade training and equipment, and protection of safe shutdown equipment. Fulfillment of these requirements resulted in physical modifications to the plant, including installation of an alternate shutdown (ASDS) panel, re-routing of safe shutdown cables, and upgrading of fire barriers. The Nuclear Regulatory Commission's (NRC) Inspection Report 50-263/86008, dated December 3, 1986, documented the satisfactory resolution of the sections of 10 CFR 50, Appendix R applicable to Monticello.

B.1.3 Overall Methodology

The Monticello fire study uses an approach that combines the deterministic evaluation techniques from the FIVE methodology with classical PRA techniques. The FIVE methodology provides a means of establishing fire boundaries as well as methods to evaluate the probability and the timing of damage to components located in a compartment involved in a fire. PRA techniques allow determination of compartment-specific core damage frequencies associated with fires within the various fire areas of the plant. Compartments were identified and evaluated, then quantified using the fault trees and event trees from the internal events PRA.

The transient event trees from the internal events PRA and related fault trees were used to perform the quantification. The resulting accident sequences were binned into three accident classes and subclasses, a subset of those used in the internal events PRA. These accident classes and their relative contributions are shown in Figure B.1.4-1. The contribution of specific areas to the core damage frequency is shown in Figure B.1.4-2.

B.1.4 Summary of Major Findings

The principle finding of this analysis is that there is no area in the plant in which a fire would lead directly to the inability to cool the core. Without additional random equipment failures unrelated to damage caused by the fire, core damage will not occur. As a result, this study concludes that there are no vulnerabilities due to fire events at the Monticello Nuclear Generating Station.

The core damage frequency resulting from fires is estimated to be less than 7.8E-6/year. Fire-induced core damage sequences total only a fraction of the total core damage frequency of the internal events PRA. This is consistent with the results of internal fire analyses at other sites.

It should be noted that these results include a number of conservative assumptions. For example, automatic or manual fire suppression was not credited except in the control room, cable spreading room and main feedwater pump area. Fires were assumed to completely engulf a sub-area once ignited. Further, repair activities were only applied to accident sequences in which a long time was available to effect repairs (on the order of 6-hrs or more) and then only to those components not damaged by the fire. When repair actions were credited, the repair of only a single failed component was assumed even if there were multiple failures to which recovery could be applied.

From Figure B.1.4.1, the core damage frequency is spread across all three accident classes; core melt with the reactor at high pressure (Class 1A), core melt at low pressure (Class-1D) and containment decay heat removal failure (Class 2).

High pressure injection systems (feedwater and HPCI, for example) and battery chargers (for long term operation of SRVs) are susceptible to damage from a fire in any of several locations. A majority of accident Class 1A results from such dependencies. Examples of these types of areas are the MCC 133/feedwater pump area and the turbine building 931' area. Alternate high pressure injection systems such as RCIC can provide adequate makeup for these scenarios, limiting their overall contribution to core damage. Also contributing to this accident class are fire sequences requiring operation from the alternate shutdown panel, such as fires in the control room and cable spreading room.

Core damage at low reactor pressure occurs following failure of high pressure injection sources and subsequent successful depressurization. Fires that dominate this accident class (Class 1D), damage multiple high and low pressure injection sources. Examples of these types of fire areas are the MCC 133/feedwater pump area and the turbine building 931' area.

Accident Class 2 includes slowly evolving accident scenarios in which decay heat removal from the containment is assumed to be lost (e.g., RHR and hard pipe vent failure). A gradual heatup of the suppression pool to saturation is assumed to occur for these events, followed by a slow pressurization of containment if RHR is not restored. More than a day into the event, the SRVs are assumed to close as a result of the low differential pressure between the containment and the pneumatic supply to the SRVs. Low pressure injection makeup would become unable to inject as the reactor repressurized. Makeup systems still capable of maintaining reactor inventory include feedwater, CRD and HPCI. Fire areas which dominate this accident class include those in which one or more of these systems are affected.

Eighty three percent of the plant risk associated with internal fires can be traced to seven rooms/burn sequences. These rooms/burn areas are (1) the main control room, (2) turbine building 931'(fire zones XII/17, 19A and 19B), (3) the MCC 133/feedwater pump area, (4) the cable spreading room, (5) the reactor building 935/962' west, (6) the lower 4KV switchgear room and, (7) the Division II area of the EFT building. A brief discussion of each of these areas follows, including a description of the means by which adequate core cooling can be assured even if a fire were to cause significant damage.

Control Room, Cable Spreading Room

If not suppressed by automatic or manual equipment, a fire in the control room or cable spreading room is assumed to cause loss of all equipment not controlled from the alternate shutdown system panel (ASDS). The ASDS panel assures the ability to shut down the plant and cool the core/containment in the event of a fire in either of these areas. Systems available at the ASDS panel include Division II of core spray, RHR (suppression pool cooling), and four of the eight SRVs. Operation of the hard pipe vent and #12 diesel generator is also possible from the ASDS panel. The availability of these systems from the ASDS panel limits the risk significance of fires in the control room and cable spreading area.

931' Turbine Building Area

A fire in this area is assumed to lead to failure of Division II low and high pressure systems. RCIC, the remaining train of FW/CRD, and Division I low pressure systems are available for injection. Division I suppression pool cooling is available for containment heat removal.

MCC 133/Feedwater Pump Area

A fire in the MCC 133/feedwater pump area is assumed to lead to failure of feedwater, one train of CRD makeup, and Division I safety systems. HPCI, RCIC, the remaining train of CRD, and Division II low pressure systems are available for injection. Division II of suppression pool cooling and the hard pipe vent remain available for containment heat removal.

Reactor Building 935/962' West (BS2)

A fire in this area has the potential for disabling several important systems. Division II of RHR, SPC and, CS as well as HPCI, HPV and SDC are assumed to fail as a result of a fire in this area. Both trains of ECCS automatic start circuitry are also located in this area. Because of the significant quantity of electrical and mechanical equipment located in this area, the ignition frequency is also large. Feedwater, RCIC and Division I low pressure systems are available for injection following fires in this area.

Lower 4KV Switchgear Room

A fire affecting equipment in the lower 4KV area could cause failure of Division I safety related equipment, RCIC (following battery depletion), feedwater (loss of LC 101 and 103) and one of the CRD pumps. Equipment remaining that are independent of this area include Division II of low pressure injection systems (LPCI and core spray) and suppression pool cooling, HPCI, a CRD pump, the SRVs and the hard pipe vent.

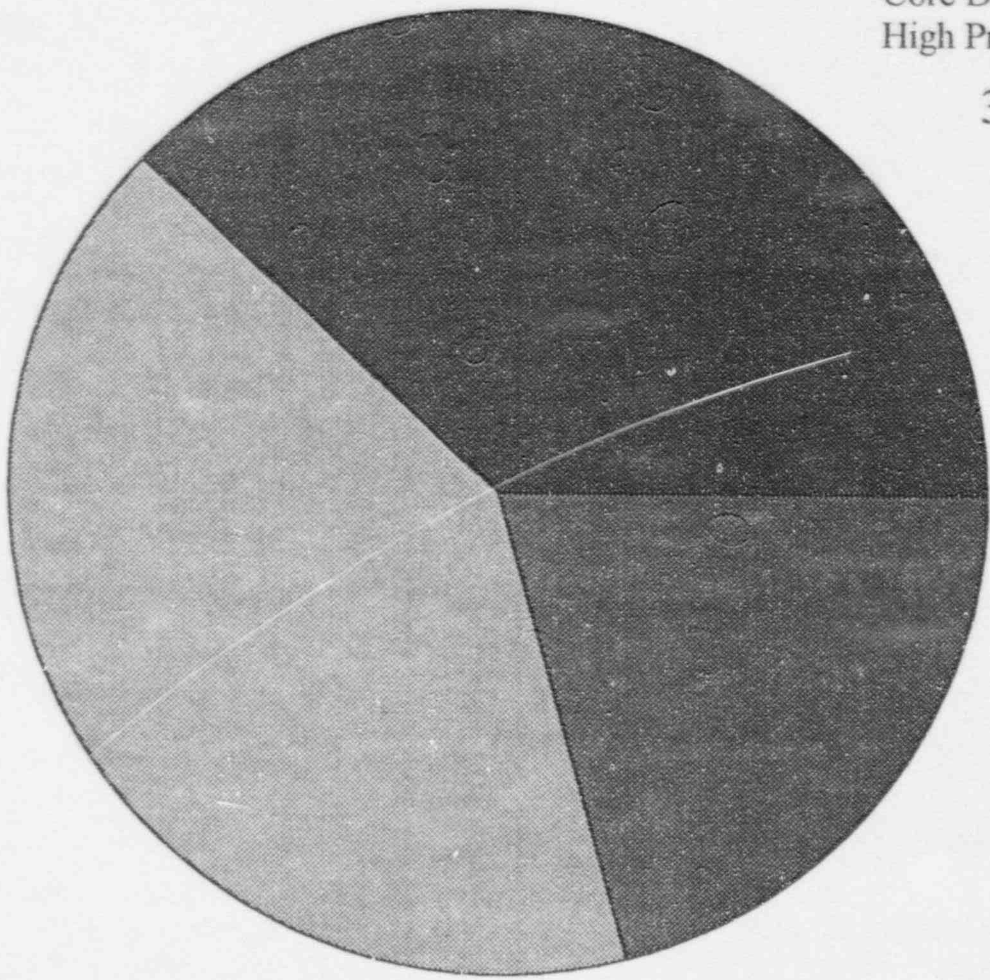
Division II Area of the EFT Building

This area contains cabling for the HPCI battery, HPV, MCC-44 and, Division II low pressure systems. 125VDC panel #211 cabling is also located in this area and provides breaker control power for most Division II equipment. Systems available for injection include RCIC and one train each of Feedwater, LPCI and Core Spray (CRD was not credited). Many of the Division II cables are located in this area because of routing to the ASDS panel located in an adjoining room.

Figure [illegible]-1

Monticello Fire PRA

(Core Damage by Accident Class)

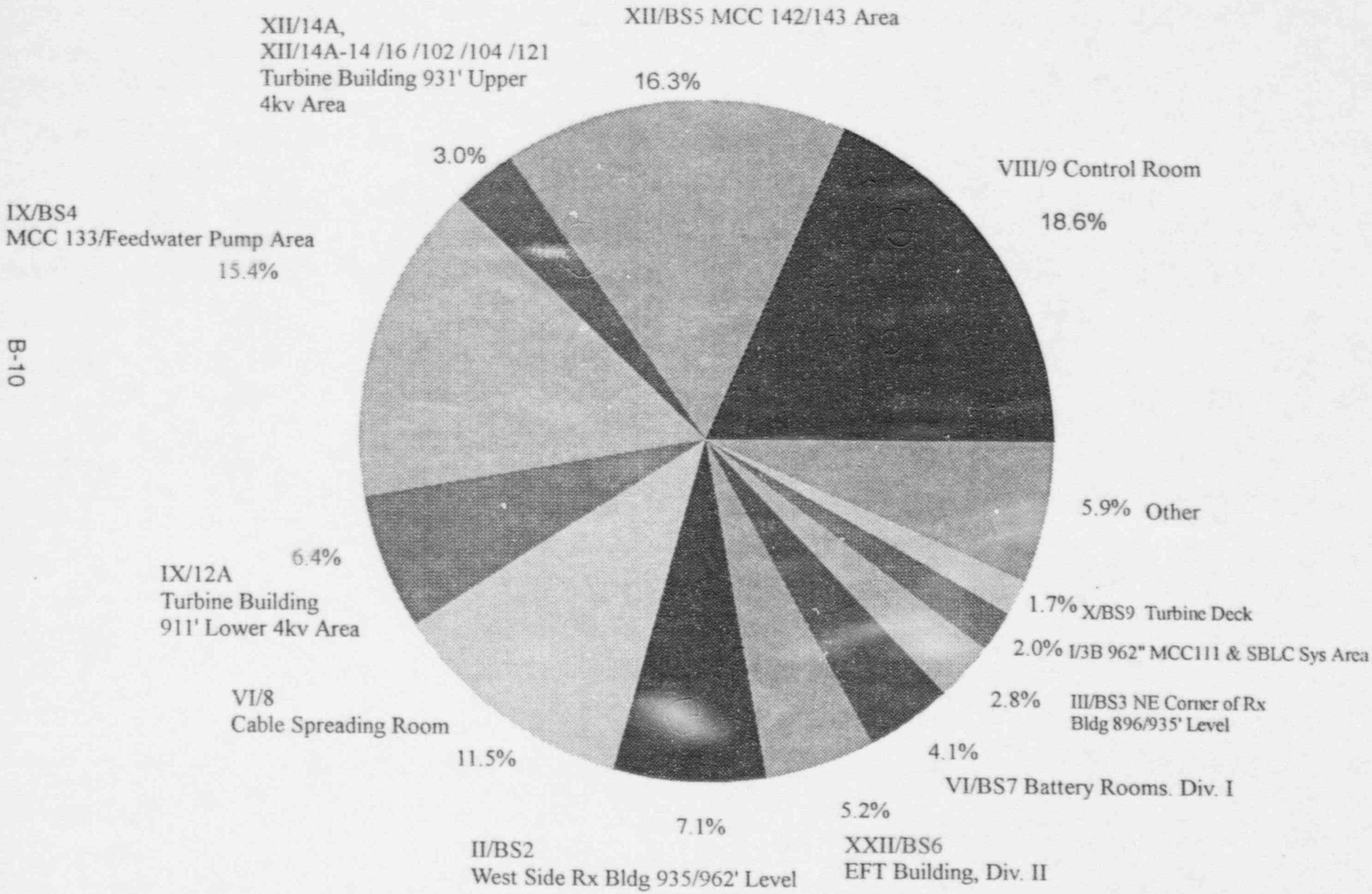


Core Damage Frequency = 7.8E-6 per year

Figure 1.4-2

Monticello Fire PRA

(Core Damage by Fire Area)



Core Damage Frequency = 7.8E-6 ar

B.2 INTERNAL FIRE ANALYSIS

B.2.1 Fire Analysis Methodology

This fire analysis combines the deterministic evaluation techniques of the FIVE methodology with classical PRA techniques. The flow chart in Figure B.2.1.1 illustrates the process used to quantify accident sequences for the Monticello fire IPEEE. Phase I is a deterministic evaluation of fire spread and ignition source frequencies. Phase II is a probabilistic evaluation of core damage using PRA techniques. If conditional core damage frequencies are unacceptable, Phase II continues with a deterministic evaluation of the effects of fire suppression and fire propagation. The FIVE methodology is used to establish fire boundaries and to evaluate the probability and the timing of damage to components located in a compartment involved in a fire. PRA techniques are used to determine compartment-specific core damage frequencies for fires within specific fire areas.

Fire areas: The Appendix R fire areas for Monticello are defined in Table B.2.1.1. For this IPEEE fire analysis, those areas outside the main reactor/turbine building complex which meet the following criteria were screened from further consideration:

1) The area contains no system credited in the internal events PRA or cables supporting those systems, and

2) A fire in the area would cause no demand for safe shutdown functions because the operating crew can maintain normal plant operations.

In applying these criteria, only fire areas outside the main reactor/turbine building complex were screened from further evaluation.

Spread of fires across boundaries: The spread of fires across fire area boundaries is addressed in the FIVE methodology. The following criteria were used to identify boundaries which can be considered to prevent the spread of a fire:

1) Boundaries between two zones, neither of which contain safe shutdown components nor plant trip initiators on the basis that a fire involving both zones would have no adverse effect on safe shutdown capability.

2) Boundaries that consist of a 2-hour or 3-hour rated fire barrier on the basis of fire barrier effectiveness.

3) Boundaries that consist of a 1-hour rated fire barrier with a combustible loading in the exposing zone of less than 80,000 Btu/ft^2 on the basis of fire barrier effectiveness and low combustible loading.

4) Boundaries where the exposing zone has very low combustible loading (<20,000 Btu/ft^2), on the basis that manual suppression will prevent fire spread to the adjacent zone.

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

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

The first criterion was not applied to the Monticello Fire IPEEE. That is, the potential for a fire to spread was evaluated whether or not there was safe shutdown equipment or plant trip initiators in a given zone.

If any one of criteria 2, 3 or 5 were met, the potential for fire spread through or across the common boundary was assumed to be negligible. These three criteria credit fire boundary ratings and combustible loading.

Criteria 4 and 6, in which fire suppression is credited, were not initially applied, to allow future evaluation of the impact of suppression and because the probability of automatic fire suppression systems failing to actuate is non-negligible. If any of the compartment fire events led to dominant core damage sequences, the effect of fire suppression was then evaluated in a probabilistic manner. This approach allowed identification of fire suppression systems that have the greatest impact on fire-induced core damage.

The groupings of fire compartments due to fire spread potential are presented in section B.2.5 and shown in Table B.2.6.3.

Systems credited: Before fire sequence quantification could be performed, it was necessary to identify the functions and systems to be included in the fire IPEEE. The associated equipment and cables and respective locations were then identified using plant documents (see section B.2.3) in conjunction with the Monticello internal events PRA and a plant walkdown. Initially, only core spray, the suppression pool cooling mode of RHR, SRV's, HPCI and RCIC systems, and their support systems were credited in the analysis because detailed information was readily available on cable location and routing for these systems [3,7]. When it became clear that other systems would greatly reduce the potential for core damage in certain sequences, those systems were credited after verifying that their cables did not run through the burning area. For example, feedwater was credited for fires in the reactor building. Feedwater and CRD were credited for fires in the turbine building only when it could be shown that they were not damaged by the fire. The hard pipe vent was credited for fire scenarios throughout the plant where it could be shown that it would not be damaged by a fire in the area.

Accident sequence evaluation: The next phase of the analysis was a multi-step, progressive probabilistic evaluation that considered the sequence of events that must occur to create the loss of safe shutdown/risk-significant functions. Figure B.2.1.1 shows the flow path and the major steps in the process. These steps consist of determining ignition source frequencies and quantifying specific fire scenarios. Following accident sequence quantification, the impact of fire suppression and the potential for the fire to propagate to identified targets was considered for risk significant areas. The potential impact on containment performance and isolation was evaluated following the core damage assessment.

The first step of the accident sequence was to identify and tally the ignition source frequencies in each compartment. These sources were identified during the first walkdown and a compartment-specific ignition frequency was calculated in accordance with the methods detailed in FIVE. Section B.2.6 details the actual methodology used in these calculations.

The next step, quantifying specific fire scenarios, was performed using the ignition source information in conjunction with the fire spread and fire effects information developed in Phase I. All the basic events in the logic models of the internal events PRA related to cables or components in the burning location were assumed to be failed. At this point in the evaluation, it was assumed that all equipment and cabling within the affected fire area or sub-area was destroyed. The core damage frequency for each of the fire areas and sub-areas was then quantified using the internal events PRA fault tree and event tree models. Fires in the control room and cable spreading room included additional actions and assumptions that were incorporated into event trees developed explicitly for these rooms. The quantification yielded a core damage frequency (CDF) for each area by incorporating the compartment-specific ignition frequencies and crediting the unaffected systems or trains included in the internal events PRA.

An additional step evaluated each of the areas whose quantification resulted in a significant contribution to overall CDF, to realistically consider the likelihood of fire-induced damage and successful fire suppression. The control room, cable spreading room and upper 4KV area (fire zone XII/14A) are comprised mainly of electrical cabinets/panels. Because these areas initially accounted for a significant portion of the total plant risk due to internal fires, fire spread between cabinets in these areas was selected for a more detailed evaluation. This evaluation resulted in the creation of subcompartments or "virtual rooms" within these areas. Based on the configuration of these enclosures and the results of experiments with electrical cabinet fires [5], fire spread beyond the enclosed metal cabinets in which the fire started was not considered credible. Switchgear located in the upper 4KV area and panels in the control/cable spreading areas were considered virtual rooms with their own ignition source frequencies and affected equipment. Load center 102, load center 104, motor control center 121, Bus 14 switchgear, and Bus 16 switchgear were considered as virtual rooms within the upper 4KV area. Individual control/instrument panels within the control/cable spreading areas were also considered virtual rooms.

Three areas, the cable spreading room, control room and feedwater pump area, were also selected for detailed fire suppression analysis. The cable spreading room was selected for more detailed analysis because it was a significant contributor to overall core damage and it is protected by an automatic suppression system. Similarly, the control room was selected for detailed analysis because it also was found to be a significant contributor to core damage and it is continuously occupied. The feedwater pump area was selected because automatic suppression systems covered the feedwater pumps and TG oil areas; both large ignition sources.

The final step was to evaluate the impact of the fires on the containment, structurally and functionally. Containment structural evaluations included factors such as combustible loading in and around the containment and the fact that the containment is inerted during power operation. The potential for containment isolation or bypass was also investigated. Most containment isolation valves fail in a safe (closed) position. Multiple failures are required to bypass the containment. Because of these and other factors, containment integrity is expected to be maintained following any postulated fire. A more detailed description of these analyses is contained in Section B.2.12.

Uncertainties: Most of the uncertainty in the results is centered around assumptions made in the accident sequence quantification. These assumptions include those regarding credit for various systems and operator actions that may occur in response to a fire as well as those implicit in the deterministic evaluation of plant response to a fire such as that contained in the FIVE methodology or experimental studies.

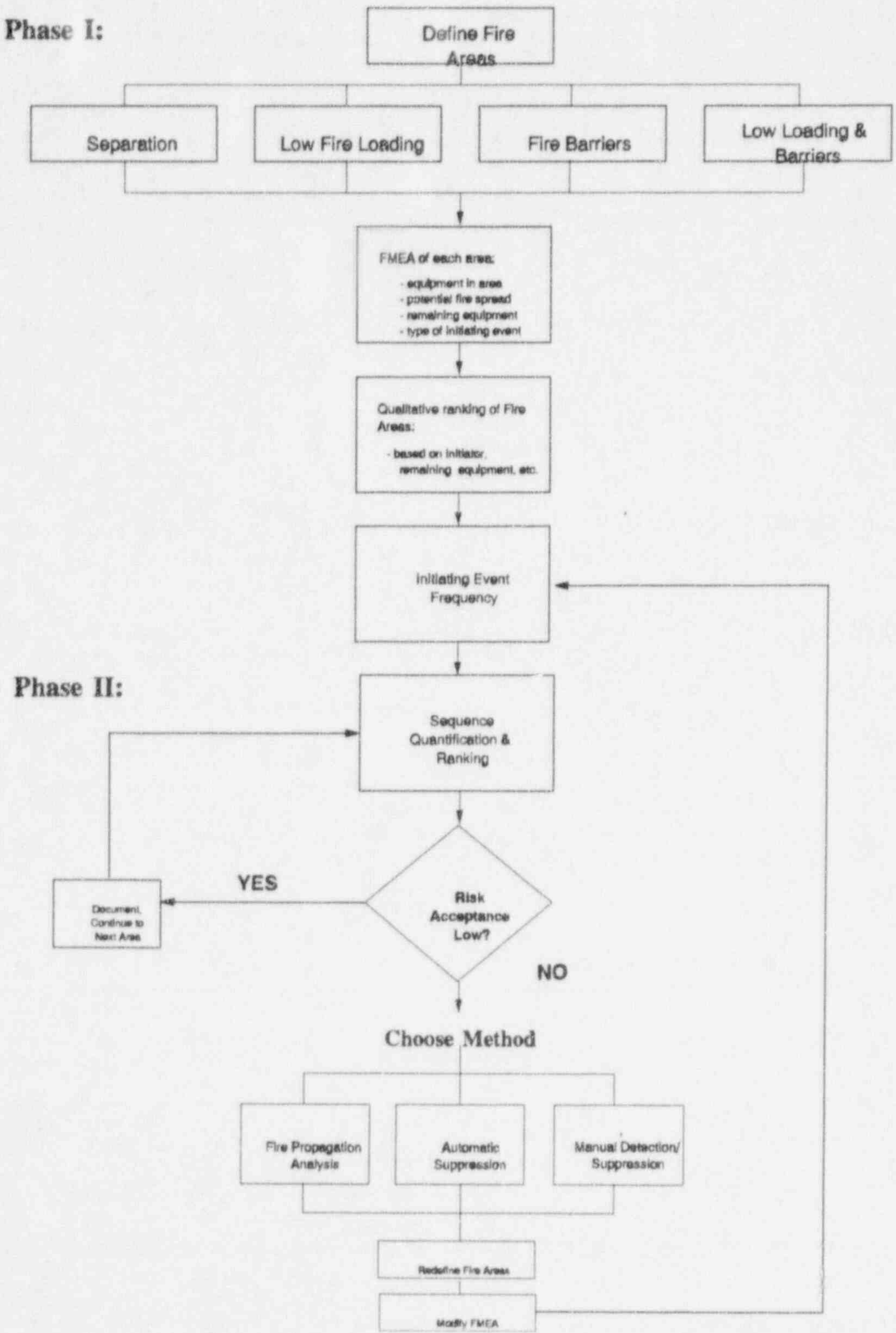
As examples, automatic and manual fire suppression were not credited except in the control room, cable spreading room and, feedwater pump area. Except in the areas where "virtual rooms" were developed, fires were assumed to completely engulf the area in which they started. If deterministic methods had been applied to show the limit of the fire spread, core damage may have been reduced. Further, repair activities were only applied to accident sequences in which a very long time was available to effect repairs, and then only to those components not damaged by the fire. When repair actions were credited, the recovery of only a single failed component was assumed even if there were multiple failures to which recovery could be applied. Systems were also assumed to fail due to fires in certain areas in order to limit the effort required to perform cable tracking. Wherever possible, assumptions such as these were made in a conservative manner to bound uncertainties.

Assumptions incorporated into risk-specific areas within the plant include the likelihood of fire propagation along horizontal cable trays (cable tunnels - fire zones 16/17) and fire spread between electrical cabinets/panels in the upper 4KV, control and cable spreading rooms. While there may be uncertainties associated with these assumptions, their application is supported by deterministic or experimental evidence under specific conditions. Further, the overall conclusions of the fire IPEEE can be shown to be insensitive to these particular uncertainties. That is, there is no one area in the Monticello plant in which a fire could start that does not require additional failures unrelated to the fire before inadequate core cooling would result.

Table B.2.1.1 Monticello Appendix R Fire Areas

FIRE AREA	DESCRIPTION
I	Reactor Building, Radwaste Building
II	Reactor Building
III	NE Corner Reactor Building, 896'-3" and 935'
IV	Reactor Building Suppression Pool Area 896'-3"
V	Reactor Building Recirc Pumps MG Set Room
VI	Admin Building D-1, D-3, D-5 Battery Rooms
VII	Admin Building D-2, D-4 Battery Room
VIII	Control Room
IX	Turbine Building - Division I Spaces
X	Turbine Building - Turbine Deck/Misc areas
XI	Diesel Fuel Oil Pump House
XII	Turbine Building - Division II Spaces
XIII	Standby Diesel Generator #12 Room 931'
XIV	Standby Diesel Generator #11 Room 931'
XV	Standby Diesel Generator Building Day Tank Room 931'
XVI	Standby Diesel Generator Building Day Tank Room 931'
XVII	Discharge Structure Pump Room
XVIII	Offgas Stack and Retention Building
XIX	Guard House
XX	Security Diesel Building
XXI	Emergency Filtration Building - Division I
XXII	Emergency Filtration Building - Division II
XXIII	Diesel Fire Pump Room
YARD	Turbine Building Extension (Transformers) 931'

Figure B.2.1.1 Fire PRA Flow Chart



B.2.2 Modeling Assumptions

The following key assumptions were made in this analysis:

1. A loss of offsite power initiating event was found not to be likely for a fire in any area. An engineering analysis [6] prepared in support of Appendix R also concludes that fire spread between transformers and between the transformers and the turbine building is not credible. This conclusion was based in part on the following: (1) the 1R and 2R transformers are separated by a block wall barrier with a three-hour fire rating, (2) the 1R and 2R transformers are separated by several hundred feet from the 1AR transformer, (3) the 1AR transformer is located a considerable distance from any main structure, and (4) automatic detection is available in conjunction with a manually-actuated deluge system. In addition, the power supplies from the exterior transformers to the Division I buses (room 12A) and the Division II buses (room 14A) are not susceptible to a single fire.

2. Feedwater was considered to be available for all fires located in the reactor building unless it failed due to random causes not related to the fire. An investigation of feedwater system components located in the reactor building indicated that no single fire in the reactor building could cause feedwater system failure.

3. Pertinent cables associated with the core spray (CS), residual heat removal (RHR), safety/relief valves (SRVs), high pressure coolant injection (HPCI), and reactor core isolation cooling (RCIC) systems were tracked throughout the plant. These systems were assumed to fail only due to non-fire-related random causes if their cables were not located in the compartment impacted by the fire.

4. The impact of fires on plant risk was quantified using the internal events PRA general transient event tree model. This event tree was selected because it most closely represented the plant response given the systems being modeled. The injection systems contained in the internal events PRA that were credited in the fire quantification were HPCI, RCIC, condensate, feedwater, core spray, RHR (LPCI mode), the fire system, and CRD, depending on the effects of the fire and knowledge of the location of cables for these systems within each fire compartment. Decay heat removal functions considered were the shutdown cooling and suppression pool cooling modes of RHR, and the hard pipe vent.

5. ATWS events are not expected to be induced by fires, due to the fail-safe design of the reactor protection system, and simultaneous occurrence of an ATWS and a fire is probabilistically insignificant.

6. LOCAs are not expected to be induced by a fire and simultaneous occurrence of a LOCA during a fire is probabilistically insignificant.

7. Reference 10 indicates that fire spread in horizontal cable trays located in cable tunnels or corridors is negligible. If the only combustible material located in the area is the cable insulation, fires in horizontal cable trays are normally self-extinguishing. For those cable fires that did not immediately self-extinguish, the maximum propagation length was approximately seven feet. It was therefore assumed that fire would not propagate the entire length of fire zone 17 prior to extinguishing itself or being suppressed.

8. The control, cable spreading and upper 4KV rooms were broken up into sub-compartments based on the low potential for electrical cabinet/panel fires to propagate beyond the cabinets in which they are located. Reference 5 provides experimental evidence supporting this assumption.

9. Fires are assumed to spread until they engulf the entire sub-area in which they start unless the fire is suppressed. No credit was taken for suppression except in the control/cable spreading rooms and feedwater pump area.

B.2.3 Review of Plant Information and Sources

Several sources of information were reviewed and used in support of the Monticello fire IPEEE. The information sources most often consulted were the Monticello individual plant examination (IPE) [2], the Monticello Updated Fire Hazards Analysis [7], the Monticello Safe Shutdown Analysis Engineering Report [3], and the Fire-Induced Vulnerability Evaluation (FIVE) Report [4]. A complete list of the references used in support of this project is contained in Section B.2.16.

The IPE was used to identify important systems and functions and provided the base fault trees and event trees used to quantify the fire-related plant risk. The IPE also provided detailed information on support systems for the important front-line systems.

The Updated Fire Hazards Analysis provided information on combustible loading, detection and suppression capabilities, and fire barrier ratings for fire areas and zones within the plant. This document also provided floor plans showing each fire barrier and identified adjacent and adjoining fire areas. The fire area/zone interaction analysis used the information contained in this document. The floor plans contained in the Updated Fire Hazards Analysis which were useful in the IPEEE are included in this appendix as Attachment 1.

Monticello system functional block diagrams and existing 10CFR50 Appendix R data were reviewed to determine the cables necessary for operation of the components included in the models. The information from the block diagrams was used to verify and to supplement the cables identified in the Safe Shutdown Analysis Engineering Report. These documents were the primary sources used to identify the cables that required tracking.

The Monticello Cabletrack database was then used to identify each of the cable trays and conduits containing these cables. The cable trays and conduits were then related to specific fire zones using the "Conduit and Trays" drawings. The result of this research was a database that contained information about cable and component locations and linked this information to basic events contained in the Monticello IPE.

This information was then used to develop a spatial database to aid in identifying, for example, all equipment impacted by a fire in a given area, or all cables and components associated with a particular event failure.

Key information contained within these records include:

- Equipment/cable ID #
- Related IPE event (relates equipment/cable to basic or developed event)
- Equipment location (for actual components)
- Cable location (fire area/zone)
- System designator
- Cable raceway designator
- Comments

B.2.4 Plant Walkdown

Various walkdowns were performed in support of the IPEEE analyses. The fire IPEEE walkdown members included a senior reactor operator, a fire protection engineer, an electrical system engineer, and a fire IPEEE analyst. A senior reactor operator and a fire IPEEE analyst participated in all walkdowns. Fire protection and electrical system engineers were called upon as questions arose, principally during the final walkdown, to confirm key assumptions used in the IPEEE.

B.2.4.1 Initial Plant Walkdown

B.2.4.1.1 Objectives of Initial Plant Walkdown

The primary objectives of the first walkdown were to gather data, confirm information and assumptions, and complete the NUREG/CR-5088 "Fire Risk Scoping Study" evaluation [11]. The walkdown was used to determine whether the assumptions and calculations, particularly fire barrier effectiveness assumptions, can actually be supported by the physical conditions that exist. This included verifying and validating (1) the combustible loading estimates in the fire hazards analysis, (2) the existence of fire protection systems, (3) fire barrier status, (4) interaction of fire areas and zones, and (5) determination of ignition sources.

B.2.4.1.2 Initial Walkdown Process

During the walkdown, a pre-printed data sheet for each compartment was completed. This data sheet contained blanks for the number and type of ignition sources located in each compartment. The sheets also contained general comment sections where the analyst noted any unique or unexpected features (combustible loading, smoke paths, fire barrier status, etc.) that could impact the analysis.

B.2.4.1.3 Findings from Initial Plant Walkdown

Several general findings were made during the walkdown. Boundary ratings were found to be generally conservative due to lack of combustible loading in close proximity to the barriers. The general condition of the plant was clean and well kept. If a compartment presented a significant ALARA concern and the area could not be inspected from the outside, the compartment was not inspected. Instead, plant documents and operator knowledge formed the basis for analysis of these spaces. Monticello's plant photodocumentation system (a digital pictorial data base of plant areas which normally have restricted access) was used to verify this information. These spaces and general information, other than ignition sources, are identified and documented in a walkdown summary.

B.2.4.2 Final Plant Walkdown

B.2.4.2.1 Objectives of Final Plant Walkdown

The objective of the final plant walkdown was to perform a confirmation of the assumptions and conclusions of the fire IPEEE. This included a detailed inspection of the barriers surrounding each of the significant fire areas and verification of the location of important cables contained within these areas. The Cabletrack database, used to identify cable routing and locations, was also checked to ensure that it was maintained up-to-date.

B.2.4.2.2 Final Walkdown Process

Important areas of the plant, as determined by the results of the fire IPEEE quantification, and areas of the plant that required validation of assumptions made during the analysis were inspected during the final walkdown. These areas included both upper and lower 4KV areas, the feedwater pump area, and portions of the EFT building including the ASDS panel. Potential fire spread paths, equipment orientation, and fire barriers were inspected.

B.2.4.2.3 Findings from Final Plant Walkdown

The following findings were generated during the final plant walkdown. As part of the resolution of NUREG/CR-5088 "Fire Risk Scoping Study" issues, inadvertent operation of a fire suppression system that could disable both trains of a system was investigated. The only system identified in which operation of the fire system may disable multiple trains of equipment is the feedwater system. It was noted that the deluge system located over the #12 feedwater pump has a fusible link installed to prevent feedwater pump damage due to inadvertent operation of the deluge system.

Even though fire zones XXII/32B and 33 are located in the same fire area, it was desirable to analyze them separately because the cables they contain affect different equipment. Because of the low combustible loading and the intervening barrier, it was assumed (based on FIVE criteria) that fire would not spread between these two zones. To validate this assumption, a walkdown of the two zones was performed. It was noted during the walkdown that all penetrations between fire zones 33 and 32B were sealed and that combustible levels were in accordance with the fire hazards analysis.

Potential for fire spread along the entire length of the turbine building cable tunnels (zones 16 and 17) was determined to be negligible based on results of full-scale fire tests performed at Fermi National Accelerator Laboratory [10] and engineering judgement. A walkdown of these areas was performed as part of the validation of this assumption. It was noted that cables are fire-stopped where they pass from the lower 4KV room to their respective cable tunnels, while cables passing from the upper 4KV area to their respective cable tunnels are not fire-stopped. Cable trays in the upper cable tunnel are lightly loaded (approximate 1/4 full), with one of the two tray sets incorporating a metal cap the entire length.

A single fire that disables all offsite power sources could have a significant impact on plant risk. For this reason, a walkdown of the turbine building 4KV rooms was performed to ensure that this scenario was not possible. It was noted that power feeds (bus ducts) from the 1R/2R transformers enter the turbine building on opposite sides of the room in the upper 4KV area. Power feeds from the 1AR transformer enter the turbine building in the lower 4KV area. This arrangement ensures a single fire will not damage all offsite power feeds.

A review was performed to assure the reliability of the database that was used in cable identification. Administrative cable and raceway controls are in effect that provide a systematic way of tracking and entering all cable modifications and data changes occurring in the plant. It was determined that these as well as other supporting procedures provide the controls and record-keeping necessary to ensure that the database is accurate and up-to-date.

B.2.5 Identification of Important Fire Areas/Zones

The Appendix R fire areas and zones provide the starting point for this analysis. In accordance with Appendix R requirements and the Monticello Updated Fire Hazards Analysis, a fire area is defined as a portion of a building that is separated from other areas by boundary fire barriers. Only a single division of safety equipment is allowed within a given fire area unless the redundant train is protected by additional separation requirements detailed in Appendix R of 10CFR50. Fire zones are subdivisions of fire areas in which fire suppression systems, spatial separation, and/or construction barriers combine to combat particular types of fires and help prevent their spread. A fire zone can be a single room or multiple adjoining rooms. A new term, fire "sub-area", was defined for use in the fire IPEEE to facilitate identification of the potential for fire spread between zones. The FIVE boundary evaluation criteria used to define sub-areas are similar to those originally used for defining Appendix R fire zones. These criteria include spatial separation of components and cables, combustible loading, and/or construction barriers. These criteria are found in the FIVE methodology and are discussed in Section B.2.1.

A scheme was developed to ensure consistent naming of the sub-areas The name consists of a Roman numeral representing the fire area followed by an alphanumeric string representing one or more fire zones. If the sub-area has precisely the same boundaries as a single fire zone, then the alphanumeric string is identical to the fire zone designation (i.e., VI/8 represents the cable spreading area). If there is the potential for fire spread between zones, then a collection of zones is given a single identifier representing a burn sequence (BS). For example, the potential for fire spread between the RCIC corner room and the TIP room, zones III/1C and III/2A, is designated as III/BS3. There are nine sub-areas which encompass more than one fire zone as shown in Table B.2.6.3.

The Monticello Nuclear Generating Plant has previously analyzed the rating of barriers incorporated in its Updated Fire Hazards Analysis for all the pre-existing fire areas and has determined that the barriers are adequate. Where those barriers are incorporated into "sub-areas" generated to support the IPEEE analysis, an additional screening for fire spread was performed. The results of this screening, based on applying conservative criteria contained in FIVE, are consistent with the analyses previously performed.

B.2.5.1 Reactor Building

The Appendix R analysis divides the reactor building into five fire areas: I, II, III, IV, and V. These fire areas were divided into fifteen sub-areas to facilitate detailed evaluation of the effects of fire within this structure. The fire area and sub-area boundaries and interfaces are described below (Attachment 1 contains drawings of these areas).

The lowest level of the reactor building, elevation 896′, is below grade with all exterior walls abutting soil. The turbine building abuts some of the upper sections of the north wall. The suppression pool room (area IV) is located in the center of this level and is separated from the adjoining fire areas by three-hour fire-rated or equivalent barriers. The HPCI room, CRD pump room, and floor drain tank room (area II) are located in the northwest corner of this level. The RCIC room (area III) is located in the northeast corner of this level, and the RHR and CS pump rooms (areas I and II) are located in opposite corners along the south wall. All fire areas located in this level are separated from the adjoining areas by three-hour fire-rated boundaries.

The ground level floor, elevation 935′, contains portions of three fire areas. This level is basically divided east and west with the east side being fire area I and the west side being area II. The traversing in-core probe (TIP) drive room located in the northeast corner is in area III. Areas I and II are separated by three-hour fire-rated boundaries. The turbine building (area X) abuts the north wall and the administration building (area VI) abuts the east wall. These common walls are also three-hour fire-rated boundaries.

Elevation 962′ is made up of three fire areas. Similar to the ground level floor, the east side is fire area I and the west side is fire area II. Fire area V is located in the northeast section of the level. All fire areas on this level are also separated by three-hour fire barriers. The ceiling of fire area II on this level is common with the floor of fire area I located on the level above. Some unsealed penetrations do communicate between these areas, however, an engineering analysis performed in support of Appendix R [6] indicates that fire spread between the two fire areas is not credible. The turbine building (area X) and the recirculation pump MG set room abut the north wall, and the administration building (area VI) abuts the east wall. These common walls are also three-hour fire-rated boundaries. Portions of the south wall form a common boundary with the radwaste area. Even though this wall is only rated for one hour, it is an equivalent three-hour boundary.

The 985′ level of the reactor building is comprised of two fire areas. Fire area I encompasses the entire level with the exception of the upper portion of the recirculation pump MG set room which is located in the northeast corner of the level and is in fire area V. Exterior walls are located along the south, east, and west. The north wall is a three-hour rated boundary that separates the reactor building from the turbine building. A small concrete vent chase situated along this wall is the only deviation from the three-hour rating.

Fire area I encompasses all of elevation 1001'. Since all of the walls are exterior, they are not common to any other fire area. Fire area I continues on the levels above and below this level, therefore neither the floor nor the ceiling are rated boundaries.

The top level of the reactor building is also completely encompassed by fire area I. The ceiling as well as the walls are exterior and therefore not rated.

The reactor building contains components and cabling for several of the systems credited in this study includes:

- Residual heat removal (RHR)
- Core spray (CS)
- Safety/relief valves (SRVs)
- High pressure coolant injection (HPCI)
- Reactor core isolation cooling (RCIC)
- Control rod drive CRD makeup (not credited if the fire is in the reactor building)
- Hard Pipe Vent (HPV)

Sub-area I/1B: This area consists of a single room, the southeast RHR and CS room. Because of low combustible loading, a fire is not expected to spread into or out of this room. A fire in this area could damage components and cables of Division I of RHR and CS as well as RCIC automatic initiation cables. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area I/2B: This area is a single room consisting of the east CRD hydraulic control unit area. Because of low combustible loading, a fire is not expected to spread into or out of this room. A fire in this area could damage components and cables of Division I of RHR, CS, and SRVs as well as RCIC control cables. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area I/2D: This area is comprised of the reactor building railroad car bay. This area contains no equipment that could cause a plant trip or that is necessary to safe shutdown of the plant. Because of low combustible loading in this as well as adjoining areas, a fire is not expected to spread to the reactor building. The sweep of the large metal doors between the reactor building and the railroad car bay ensures a lack of continuity of combustibles. Because of these factors, this area was removed from further analysis.

Sub-area I/2E: This area is a single room consisting of the TIP room. Because of low combustible loading in this as well as adjoining rooms, a fire is not expected to spread into or out of this room. A fire in this area could damage HPCI and RCIC control and/or power cables. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area I/2G: This area is a single room consisting of the east shutdown cooling room. Because of low combustible loading in this as well as adjoining rooms, a fire is not expected to spread into or out of this room. A fire in this area could damage cables or components vital to

Division I LPCI and the shutdown cooling mode of RHR. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area I/3B: This area is a single room consisting of a motor control center (MCC 111) and the standby liquid control system. Because of low combustible loading in this as well as adjoining spaces, a fire is not expected to spread into or out of this room. A fire in this area could damage control or power cables vital to Division I of core spray and the containment spray mode of RHR. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area I/BS8: This area consists of ten Appendix R fire zones located within the reactor building (zones 3E, 4A-E, 5A-C and 6). These zones are located in the upper levels of the reactor building and contain no cables or components related to the systems credited in this study. In addition, the combustible loading in each of these zones is such that fire spread beyond the initiating zone is not credible. However, because of the potential for a manual plant shutdown due to a fire in these zones, a core damage contribution for this area was included in the cumulative results. For simplicity, all ten fire zones have been grouped as a single fire sub-area.

Fire Area I - Zones 21A through D: These zones consist of the radwaste building and radwaste processing areas. The area contains no equipment that could cause a plant trip or that is necessary for safe shutdown of the plant. Separation from the reactor building is provided by a three-hour equivalent boundary. Because of these factors, these zones were qualitatively screened from further analysis.

Sub-area II/BS1: This area consists of three fire zones located within the reactor building (zones 1D, 1E, and 1G). These zones are located below ground level and contain only cables and components related to the HPCI and CRD systems. The intervening boundaries could not be screened based on the FIVE criteria, therefore it is assumed that a fire starting in any of these fire zones will spread and engulf all three. The fire is not expected to spread to adjoining sub-areas because of low combustible loading and lack of combustible continuity. Because of the potential for an automatic or manual plant shutdown due to a fire in this area, a core damage contribution for this area is included in the cumulative results.

Sub-area II/BS2: This sub-area consists of four Appendix R fire zones located in the west half of the reactor building, all contained within Appendix R fire area II (zones 2C, 2H, 3C, and 3D). These fire zones were combined into a single sub-area because of the potential for fire spread among them. The intervening boundaries do not meet FIVE criteria for preventing fire spread across boundaries, so it is assumed that a fire starting in any of these zones will spread and engulf all four. Cables for Division II systems, including HPCI and HPV, are found in this area. Additionally, cables for equipment controlled from the ASDS panel and cables associated with both trains of ECCS autostart circuitry also transit this area.

Sub-area II/1A: This area consists of a single room, the southwest RHR and CS pump room. Because of low combustible loading, a fire is not expected to spread into or out of this space. A fire in this area could damage components and cables of Division II of RHR and CS. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area II/2F: This area is a single room consisting of the steam chase. Because of low combustible loading in this as well as adjoining compartments, a fire is not expected to spread into or out of this room. A fire in this area could damage control and/or power cables vital to HPCI and RCIC operation. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area III/BS3: This area, identical to Appendix R fire area III, consists of two rooms located in the northeast corner of the reactor building (zones 1C and 2A). Because of an open stairwell between these rooms, fire spread between them is possible. Fire in this sub-area could fail Division II low pressure injection systems as well as RCIC. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area IV/1F: This area consists of the suppression pool room, interchangeable with Appendix R fire area IV, and is located in the below-grade portion of the reactor building. All adjoining walls and the ceiling are rated three-hour fire boundaries. The floor rests on soil. This space has an extremely low combustible loading (<1 minute fire severity). Because of low combustible loading in this as well as adjoining rooms, a fire is not expected to spread into or out of this space.

This space contains cables and components associated with HPCI, RCIC, HPV, and both trains of safe shutdown equipment. However, each train is physically separated by approximately 100 feet with no intervening combustibles. Essentially no combustible material is stored or located in the area. All cables are in conduits. The area also contains smoke detection equipment. These factors justified an exemption to Appendix R requirements and also justify the position that fire will not spread between redundant trains of safe shutdown equipment. Therefore, for the purpose of this analysis, a fire in this area is assumed to disable Division I safe shutdown equipment and the hard pipe vent. Contribution to core damage due to a fire in this area is included in the cumulative results.

Sub-area V/3A: This area is a single room, interchangeable with Appendix R fire area V, consisting of the recirculation pump MG set room. All boundaries of this room are rated at three hours with the exception of the interface with Appendix R fire zone I/4E. Fire spread into or out of this space is not expected.

The recirculation pump MG sets and cables associated with shutdown cooling valve MO-2030 are located in this area. Contribution to core damage due to a fire in this area is included in the cumulative results.

B.2.5.2 Administration/EFT Building

The administration building is located east of the reactor building and south of the turbine building, sharing a common wall with each building. The EFT building is located east of the turbine building and north of the administration building, also sharing a common wall with each building. Passage is available between the administration building and the other structures.

The Appendix R analysis divides the administration/EFT building into five fire areas (VI, VII, VIII, XXI, and XXII). These five fire areas are further subdivided into twelve fire zones. These fire zones were combined into eight sub-areas to facilitate detailed evaluation of the effects of fire within this structure. The results of the initial evaluation are described below.

Sub-area VI/BS7: This sub-area consists of two Appendix R fire zones (7A and 7B), the Division I battery rooms. A fire in either room is expected to fail Division I 125 VDC and Division I 250 VDC power.

This sub-area is separated from the turbine building by a three-hour rated boundary. The only other Appendix R fire area that adjoins this sub-area and is not separated by a full three-hour rated boundary is the Division II 125 VDC battery room. An engineering analysis [6] prepared in support of Appendix R indicates that fire spread across the boundary is not credible.

Fire Area VI Zones 10 and 11: This sub-area is comprised of Appendix R fire zones VI/10 and VI/11. There are no safe shutdown components located in the area. The items of concern are the Division II DC power cables that are routed under this area in a subterranean vault. However, this arrangement will withstand a design basis fire and is not considered susceptible to a fire in this area.

The administration building is separated from the reactor building by three-hour rated boundaries. For the most part, three-hour rated boundaries also provide separation from the turbine building. An engineering analysis [6] prepared in support of Appendix R was performed on the small section of wall (zone 10/19C interface) that does not provide a three-hour boundary. This analysis concluded that fire spread across the boundary is not credible.

The only other Appendix R fire area that adjoins this sub-area and is not separated by a full three-hour rated boundary is battery room D2 & D4 (Division II - 125 VDC). An Appendix R engineering analysis [6] was also performed on this interface. The result of the analysis indicated that fire spread across the boundary is not credible.

Sub-area VI/8: This sub-area consists of the cable spreading room. The cable spreading room is located directly underneath the control room and contains instrumentation cables for equipment monitored and controlled in the control room. In addition to cabling, the cable spreading room contains various electrical cabinets such as relay cabinets, reactor protection system distribution panels, and instrument AC panels. An unsuppressed fire in this area could potentially lead to failure of all safety-related equipment not controllable from the ASDS panel.

This sub-area is separated from the turbine building and reactor building by three-hour rated boundaries. The boundaries that are common with the remainder of fire area VI are inspected fire barriers. Structural steel members are protected and three-hour rated. This room contains automatic fire detection (smoke and thermal) and suppression (Halon 1301). Because of these features, a fire is not expected to spread into or out of the area.

Sub-area VII/7C: This sub-area consists of a single fire zone, the Division II battery room. A fire in this room is expected to fail Division II 125 VDC power.

This sub-area is separated from most administration building fire areas by three-hour rated boundaries. The only Appendix R fire area that adjoins this sub-area and is not separated by a full three-hour rated boundary is the Division I 250 VDC battery room.

Sub-area VIII/9: This sub-area consists solely of the control room and is located on the 951′ level of the administration building. This sub-area consists of a single Appendix R fire zone and is separated from other fire areas by three-hour barriers. The control room contains controls and monitoring instrumentation for most of the equipment used to achieve safe shutdown of the plant. Loss of this area due to a fire is assumed to disable all equipment that cannot be controlled from the ASDS or local panels.

The control room has battery-operated smoke detectors in cabinets and at the ceiling. These detectors sound alarms locally. The control room is continuously staffed and early detection of a fire is very likely. If the fire is not suppressed, the fire may grow to the point that control room evacuation is necessary and control from the ASDS panel is specified. These actions are detailed in Emergency Operating Procedure C.4-C, "Shutdown Outside Control Room."

Sub-area XXI/31A: This sub-area consists of one of the two fire zones (31A) that comprise Appendix R fire area XXI. The light combustible loading in conjunction with the Appendix R analysis indicate that fire spread into or out of this fire area is unlikely. These same factors also make fire spread to the other fire zone in this fire area unlikely. Because there are significant differences in the equipment in these two fire zones, they were separated into individual sub-areas.

Sub-area XXI/32A: This sub-area consists of one of the two fire zones (32A) that comprise Appendix R fire area XXI. The light combustible loading in conjunction with the Appendix R analysis indicates that fire spread into or out of this fire area is unlikely. These same factors also make fire spread to the other fire zone in this fire area unlikely. Because there are significant differences in the equipment in these two fire zones, they were separated into individual sub-areas.

Sub-area XXII/BS6: This sub-area consists of all of fire area XXII except zone 33 and includes zones 31B and 32B. Zone 33 was analyzed separately because the small number of conduit penetrations between zone 33 and this sub-area are fire-stopped. Combustible loading in these spaces is light, and there are significant differences in the equipment located in the two areas.

Sub-area XXII/33: This sub-area consists of fire zone 33. This fire zone was made a unique sub-area for the reason stated above. The ASDS panel(s) are located in this sub-area.

B.2.5.3 Turbine Building and Associated Areas

This region of the plant consists of the recombiner building, turbine building, turbine building addition, and attached structures along the northeast perimeter of the turbine building. These structures are accounted for in the Appendix R analysis by three turbine building fire areas and four fire areas associated with the diesel generators. These fire areas are further subdivided into twenty-eight fire zones. The fire area boundaries and interfaces are described below.

The lowest level of the turbine building, elevation 908′, is below grade with all exterior walls except the south wall abutting soil. The south wall, a three-hour rated concrete wall, forms a common boundary with the lower level of the reactor building. This level contains two fire areas. The rooms and corridors along the north and east perimeter are included in fire area IX with the remaining rooms contained in fire area X. With the exception of the Division I switchgear room, the same fire areas continue upward to the next level. The two fire areas located on this level are separated from each other by three-hour fire-rated boundaries or are supported by an engineering analysis [6] performed for Appendix R that indicates fire spread between the two fire areas is not credible.

The ground level of the turbine building, 931′, is comprised of three fire areas (IX, X, XII) with the majority of the level included in fire area X. The corridors and rooms located along the north and east perimeter of the turbine building make up fire area XII. The reactor building abuts the major portion of the south wall. The diesel generator structures, attached to the northwest corner of the building, are separated by three-hour fire-rated boundaries.

The highest level (turbine deck - 951′) is encompassed by a single fire area (X). This area shares a common wall with the reactor building to the south and with the administration/EFT building along the southeast corner of the building.

The turbine building fire areas/zones were combined into eight sub-areas to facilitate detailed evaluation of the effects of fire within this structure. The results of the initial evaluation are as follow:

Sub-area IX/12A: This sub-area is a single Appendix R fire zone and contains the lower 4KV area. This area is located in the northwest corner of the turbine building on the 911′ level. The upper 4KV area is located directly above this room. With the exception of the small interface with the cable tunnel, all boundaries are three-hour rated. The boundary with the cable tunnel is fire-stopped and fire spread out of or into this space is not expected.

Three large 4KV buses and three 480V load centers are located in this room. These buses distribute AC power to all Division I components. The Division I diesel generator (#11) is also aligned to one of the 4KV buses in this area as a source of backup power. Loss of this space would lead to complete or partial failure of feedwater/condensate and Division I safety systems.

Fire in this room would be detected by smoke detectors that alarm in the control room. Because of the significant amount of electrical equipment located in this space, no automatic sprinkler system is installed. Portable extinguishers and hose stations are available to combat a fire.

Sub-area IX/BS4: Sub-area IX/BS4 consists of seven Appendix R fire zones (13A-C, 16, 19C, 23B, and 37). These zones are located along the north and east perimeter of the building on the 911′ level. All adjoining fire areas are separated by three-hour rated barriers or have supporting analyses that conclude that fire spread across the boundaries is not credible. These zones were combined into a single sub-area because the combustible loading was above the level that would allow screening of the boundaries and some of the defined boundaries were not actual physical barriers.

The feedwater pumps, service/instrument air compressors, and several MCCs are located in this sub-area. In addition, several large cable trays containing various Division I power and control cables run the length of these spaces.

Automatic sprinklers are located above the feedwater pumps and above the turbine generator lube oil reservoir. Smoke detectors that alarm in the control room are located throughout the sub-area. Portable extinguishers and hose stations are also available to combat a fire.

Sub-area IX/23A: Sub-area IX/23A consists of a single Appendix R fire zone (23A). This fire zone contains a variety of pumps (RHRSW, circulating water, service water, etc.), and several MCCs and electrical cabinets. Both trains of these systems are located in this room. However, many features contribute to ensure that fire spread to both trains is not credible. The physical separation (>20′), lack of intervening combustibles (including fire stops on cable runs), and an area-wide automatic suppression system combine to prevent fire spread between trains. Because of these features, only a single train of equipment located in this space was assumed to fail in the event of a fire. The area boundaries (three-hour rated or equivalent) in addition to the low combustible loading and presence of automatic suppression prevents the spread of fire into or out of this room.

Sub-area X/BS9: The whole of Appendix R fire area X, with the exception of the recombiner building, was considered a single sub-area (i.e., zones 12B-E, 14B-C, and 30). The combustible loading in these spaces is extremely low. The major combustible contributors in this area are lube oil and seal oil, both of which are normally contained and not exposed to the atmosphere or ignition sources. Because of the low combustible loading and structural barriers, fire spread into or out of the individual rooms within this sub-area is not expected. Fire spread into or out of this fire area is prevented by three-hour rated barriers or is shown not to be credible through an engineering analysis [6] prepared in support of Appendix R requirements.

Sub-area X/22: This sub-area is a single Appendix R fire zone and consists of the recombiner building. It has extremely low combustible loading. This low combustible loading in conjunction with the relatively low combustible loading of the adjoining fire zone makes fire spread from this compartment unlikely. This area was screened from further consideration because no equipment credited in the analysis is located in this space and an immediate plant shutdown is not required if a fire were to occur.

Sub-area XII/14A: This sub-area is a single Appendix R fire zone and contains the upper 4KV area. This area is located in the northwest corner of the turbine building on the 931′ level. The lower 4KV area is located directly below this room. With the exception of the interface with the cable tunnel, all boundaries are three-hour rated. Fire is not expected to spread beyond the cable tunnel, because of its length and orientation.

Three large 4KV buses and two 480V load centers are located in this room. These buses distribute AC power to all Division II components. The Division II diesel generator (#12) is also aligned to one of the 4KV buses as a source of backup power. Loss of this space would lead to complete or partial failure of feedwater/condensate and Division II safety systems.

This room is covered with smoke detectors that alarm in the control room. Because of the significant amount of electrical equipment located in this space, no automatic sprinkler system is installed. Portable extinguishers and hose stations are available to combat a fire.

Sub-area XII/BS5: Fire zones 17, 19A and 19B were combined into a sub-area. The higher combustible loading of these fire zones and lack of distinct physical barriers (19A/19B) precluded their analysis as independent rooms. These zones are separated from other fire sub-areas by three-hour rated barriers or supported by engineering analyses [6] prepared in support of Appendix R, showing fire spread to other sub-areas is not credible.

The #14 air compressor, and several MCCs are located in these spaces. In addition, several large cable trays containing various Division II power and control cables run the length of these spaces.

Fire area XII - Zones 18A&B, 20, 34, 35, and 36: This sub-area consists of the fire area XII fire zones located outside the main turbine building complex. These zones include the machine shop, the auxiliary boiler room and the diesel generator #13 rooms. The structural walls that separate these zones from the main turbine building are three-hour equivalent barriers. These zones were combined into a single area because they contain no safe shutdown equipment and because fire spread to the turbine building is not anticipated.

B.2.5.4 Other Fire Areas

Appendix R Fire areas XIII/XIV/XV/XVI: These fire areas contain the emergency diesel generators and their respective fuel oil day tanks. These rooms are attached to the northwest corner of the turbine building. Each room is a separate fire area consisting of a single fire zone, and all boundaries are rated for three hours, except the wall between the day tank rooms which is rated for two hours.

A fire in any of these rooms will not spread beyond that room and subsequently will not impact the operation of the other diesel generator. Contribution to core damage due to a fire in these areas is included in the cumulative results.

Appendix R Fire areas XI/XVII/XVIII/XIX/XX/XXIII/YARD: These fire areas are not connected to the main reactor building/turbine building complex. The physical separation of the buildings preclude fire spread to the main complex. In addition, with the exception of the YARD transformers, there is no equipment in these fire areas that will impact the safe shutdown of the plant. As for the YARD transformers, an engineering analysis [6] prepared in support of Appendix R shows that fire spread to the turbine building is not credible. Because of these factors, all these areas were qualitatively screened from further consideration. Contribution to core damage due to a fire in these areas was not included in the cumulative results.

B.2.6 Fire Ignition Data

It is necessary to calculate ignition source frequencies for each area or zone to allow quantification of the impact of a fire in each sub-area. These individual impacts can be summed to yield the impact to the plant from all fires.

The EPRI "Fire Event Database for U.S. Nuclear Power Plants" [8] was used to estimate fire ignition source frequencies for all rooms in the plant. This database contains a total of 800 events during a period from 1965-1988. These events were compiled from 114 BWR and PWR units across the United States, representing a total sample of approximately 1300 reactor years of operation. The data includes fire incidents caused by both fixed and transient sources due to normal operations and maintenance activities.

FIVE incorporated this information into a procedure to develop ignition source frequencies for individual fire areas and sub-areas. This process was used to evaluate the ignition frequencies (F_i) for each fire compartment. An ignition source data sheet was completed for each room or fire compartment that was contained within the fire sub-areas defined in Phase I.

The four-step process identified in the FIVE methodology was used to develop the ignition source data sheet. The first step requires that the location which corresponds best to the fire compartment in question be selected. Some locations may be specific Appendix R fire areas, such as the control room and cable spreading room, while other locations may be general, such as turbine building fire area X.

The second step requires that a location weighting factor (WF_L) be determined from this classification. The weighting factor is used to translate the generic fire frequencies, compiled in FIVE (Table B.2.6.1), for a location to specific, single-unit fire frequencies. The location weighting factors are designed to account for the relative amount of ignition sources at Monticello compared to the average nuclear power plant. These factors are easily calculated using the simple formulas found in Table B.2.6.1.

The third step requires that weighting factors for each type of ignition source (WF_{LS}) be determined. The potential ignition sources in each room were obtained from the Appendix R fire analysis, equipment drawings, and a walkdown of each compartment. The amount of cabling and electrical cabinet contribution for each fire compartment was also obtained from the Appendix R fire analysis. Some ignition sources, such as cables and transformers, are best apportioned on a plant-wide basis. Once the number of plant-wide ignition sources was identified, the WF_{LS} was determined by dividing the number of components in the room by the total number of similar components in the building or generic location being considered.

The fourth step requires that the fire compartment fire frequency (F_i) be calculated for each fire compartment. Table B.2.6.2 lists the fire frequency for each ignition source by location. F_i is the sum of the ignition source frequencies for each ignition source (F_{if}) located within the given fire compartment. This value was obtained for each fire compartment by multiplying:

1) The fire frequency (F_f) (Table B.2.6.2),
2) The weighting factor for the location (WF_L), and
3) The weighting factor for each ignition source (WF_{LS}).

$$F_{if} = F_f * WF_{LS} * WF_L$$

This calculation was repeated for each ignition source in the compartment and the total fire frequency for the specific fire compartment (F_1) was calculated as:

$$F_1 = \Sigma F_{if}.$$

The resultant ignition frequencies for each compartment are provided in Table B.2.6.3.

Table B.2.6.1 Weighting Factors for Adjusting Generic Location Fire Frequencies for Application to Plant-Specific Locations (taken from FIVE methodology)

PLANT LOCATION	WEIGHTING FACTORS[1] (WF_L)
Auxiliary Building (PWR)	The number of units per site divided by the number of buildings.
Reactor Building (BWR)[2]	The number of units per site divided by the number of buildings
Diesel Generator Room	The number of diesels divided by the number of rooms per site.
Switchgear Room	The number of units per site divided by the number of rooms per site.
Battery Room	The number of units per site divided by the number of rooms per site.
Control Room	The number of units per site divided by the number of rooms per site.
Cable Spreading Room	The number of units per site divided by the number of rooms per site.
Intake Structure	The number of units per site divided by the number of intake structures.
Turbine Building	The number of units per site divided by the number of buildings.
Radwaste Area	The number of units per site divided by the number of radwaste areas.
Transformer Yard	The number of units per site divided by the number of switchyards.
Plant-Wide Components (Cables, transformers, elevator motors, hydrogen recombiner/analyzer)	The number of units per site.

1. The analyst must identify the number of like locations when determining the number of buildings, e.g., a 480-volt load center is "like" a switchgear room.
2. Reactor building does not include containment.

Table B.2.6.2 Fire Ignition Sources and Frequencies by Plant Location

Plant Location	Fire Ignition/Fuel Source	Ignition Source Weighting Factor	Fire Frequency[1,2]
Reactor Building (BWR)[2]	Electrical cabinets Pumps	B B	5.0×10^{-2} 2.5×10^{-2}
Diesel Generator Room	Diesel generators Electrical cabinets	A A	2.6×10^{-2} 2.4×10^{-3}
Switchgear Room	Electrical cabinets	A	1.5×10^{-2}
Battery Room	Batteries	A	3.2×10^{-3}
Control Room	Electrical cabinets	A	9.5×10^{-3}
Cable Spreading Rm	Electrical cabinets	A	3.2×10^{-3}
Intake Structure	Electrical cabinets Fire Pumps Others	A A A	2.4×10^{-3} 4.0×10^{-3} 3.2×10^{-3}
Turbine Building	T/G Excitor T/G Oil T/G Hydrogen Electrical cabinets Other pumps Main feedwater pumps Boiler	B B B B B A B	4.0×10^{-3} 1.3×10^{-2} 5.5×10^{-3} 1.3×10^{-2} 6.3×10^{-3} 4.0×10^{-3} 1.6×10^{-3}
Radwaste Area	Miscellaneous components	A	8.7×10^{-3}
Transformer Yard	Yard xfmers (spread to TB) Yard xfmers (LOSP) Yard transformers (Others)	A A F	4.0×10^{-3} 1.6×10^{-3} 1.5×10^{-2}

Table B.2.6.2 (Continued) Fire Ignition Sources and Frequencies by Plant Location

Plant Location	Fire Ignition/Fuel Source	Ignition Source Weighting Factor	Fire Frequency[1,2]
Plant-Wide Components	Fire protection panels	F	2.4×10^{-3}
	RPS MG sets	F	5.5×10^{-3}
	Non-qualified cable run	E	6.3×10^{-3}
	Junction in non-qualified cable	E	1.6×10^{-3}
	Junction box in qualified cable	E	1.6×10^{-3}
	Transformers	F	7.9×10^{-3}
	Battery chargers	F	4.0×10^{-3}
	Off-gas/H_2 Recombiner (BWR)	G	8.6×10^{-2}
	Hydrogen Tanks	G	3.2×10^{-3}
	Misc. hydrogen fires	C	3.2×10^{-3}
	Gas turbines	G	3.1×10^{-2} [4]
	Air compressors	F	4.7×10^{-3}
	Ventilation subsystems	F	9.5×10^{-3}
	Elevator motors	F	6.3×10^{-3}
	Dryers	F	8.7×10^{-3}
	Transients	D	1.3×10^{-3} [3]
	Cable fires caused by welding	C	5.1×10^{-2} [3]
	Transient fires due to welding/cutting	C	3.1×10^{-2} [3]

1. Frequencies are per reactor year unless otherwise noted.
2. Fire frequencies are per fraction of ignition sources per year.
3. Fire frequency represents one event. The thirteen transient events which occurred during power operation are considered by the weighting factor.
4. Fire frequency represents an estimated 130 gas-turbine-operating years.

Table B.2.6.2 (Continued) Fire Ignition Sources and Frequencies by Plant Location

Notes for Ignition Source Weighting Factor Method:

Zone specific ignition sources were determined during the initial walkdown. Normally, ignition source frequencies are estimated using methods other than direct counting, including engineering judgement. These estimates are then verified during the walkdown. Estimates should be within 25% of actual values.

A. No ignition source weighting factor is necessary.

B. Obtain the ignition source weighting factor by dividing the number of ignition sources in the fire compartment by the number in the selected location.

C. Obtain the ignition source weighting factor by calculating the inverse of the number of compartments in the locations. Exclude any areas contained in locations other than in this table.

D. Obtain the ignition source weighting factor by summing the factors for ignition sources which are allowed in the zone and divide by the number of zones in the locations in this table. For example, if cigarette smoking is prohibited do not include the cigarette smoking factor in the calculation. The factors are:

• Cigarette Smoking	2
• Extension Cord	4
• Heater	3
• Candle	1
• Overheating	2
• Hot Pipe	1

Overheating addresses errors while heating potential combustibles, e.g., battery terminal grease.

E. Obtain the ignition source weighting factor by dividing the weight (or BTUs) of cable insulation in the area by the total weight (or BTUs) of cable insulation in Appendix R fire areas, not including the fire areas in either the radwaste area or the containment. Cable insulation weights (or BTUs) are provided in Appendix R combustible loadings. (Junction boxes and splices are assumed to be distributed in proportion to the amount of cable.)

F. Obtain the ignition source weighting factor by dividing the number of ignition sources in the fire area by the total number in all the locations in this table.

G. Obtain the ignition source weighting factor by dividing the number of ignition sources in the fire area by the total number in all plant locations, include locations that were not specified in this table.

Table B.2.6.3 Monticello Ignition Source Frequencies and IPEEE Fire Area/Zone Definitions

APPENDIX R FIRE AREA/ZONE	IPEEE FIRE SUB-AREA	ZONE DESCRIPTION	HEAT LOAD (Btu/ft^2)	IGNITION SOURCE FREQ. (per year)
I/1B		896′ SE RHR and CS Room - Division I	6,836	3.49E-3
I/2B		935′ CRD Hydraulic Control Area	17,620	5.74E-3
I/2D	(Screened)	935′ SE Reactor Bldg Railroad Car Bay	11,342	3.30E-4
I/2E		935′ NE T.I.P. Room	11,172	1.00E-4
I/2G		East Shutdown Cooling Room	1,022	1.00E-4
I/3B		962′ MCC & Standby Liquid Control Sys	10,508	9.98E-3
I/3E	I/BS8 (Burn Seq. 8)	962′ Contaminated Records Storage	10,000	2.10E-4
I/4A		985′ Equipment Hatch Area	8,652	1.25E-2
I/4B		985′ Cooling Water HX Area	6,958	3.63E-3
I/4C		985′ Corridor Outside Main Exh Plenum	7,054	1.10E-4
I/4D		985′ Standby Gas Treatment Room	16,111	1.46E-3
I/4E		985′ Reactor Plenum Room	142	1.08E-3
I/5A		1001′ S Laydown and Decon Area	11,162	1.74E-3
I/5B		1001′ N Contaminated Equip Storage	1,538	2.40E-4
I/5C		985′/1001′ Skimmer Surge Tank Area	496	2.38E-3
I/6		1027′ Refueling Floor	1,294	7.53E-3
			Total for Sub-area I/BS8	3.09E-2

Table B.2.6.3 (Continued) Monticello Ignition Source Frequencies and IPEEE Fire Area/Zone Definitions

APPENDIX R FIRE AREA/ZONE	IPEEE FIRE SUB-AREA	ZONE DESCRIPTION	HEAT LOAD (Btu/ft^2)	IGNITION SOURCE FREQ. (per year)
I/21A	(Screened)	935′ Radwaste Control Room	9,736	7.14E-3
I/21B		935′ RWB Baler & Dry Waste Storage	62,840	2.30E-4
I/21C		935′ RWB Radwaste Shipping Area	89,055	5.90E-4
I/21D 935′		935′ RWB Low Level Radwaste Storage	27,359	1.36E-3
I/21D 947′		947′ RWB Radwaste Bldg	131	3.40E-4
I/21D 962′		962′ RWB SW Vent & RW Tank Area	425	3.20E-4
I/21D 985′		985′ RWB SW RW Area - Fuel Pool Waste Processing	1,203	1.00E-4
II/1D	II/BS1 (Burn Seq. 1)	896′ NW Equip & Floor Drain Tank Rm	460	3.20E-4
II/1E		896′ HPCI Room	23,540	5.74E-3
II/1G		923′ NW CRD Pump Room	11,087	2.94E-3
			Total for Sub-area II/BS1	9.00E-3
II/2C	II/BS2 (Burn Seq. 2)	935′ CRD HCU and HVAC Area	17,137	8.23E-3
II/2H		West Shutdown Cooling Room	940	1.00E-4
II/3C		962′ MCC Area	25,122	5.53E-3
II/3D		962′ Cooling Water Pump & Chiller Area	5,217	4.88E-3
			Total for Sub-area II/BS2	1.87E-2
II/1A		896′ SW RHR & CS Pump Room - Div II	6,377	3.90E-3
II/2F		935′ Steam Chase	348	6.60E-4

Table B.2.6.3 (Continued) Monticello Ignition Source Frequencies and IPEEE Fire Area/Zone Definitions

APPENDIX R FIRE AREA/ZONE	IPEEE FIRE SUB-AREA	ZONE DESCRIPTION	HEAT LOAD (Btu/ft^2)	IGNITION SOURCE FREQ. (per year)
III/1C	II/BS3 (Burn Seq. 3)	896′ NE RCIC Room	7,209	5.27E-3
III/2A		935′ NE T.I.P. Drive Room	60,674	2.74E-3
		Total for Sub-area II/BS3		8.01E-3
IV/1F		896′ Suppression Pool Area	1,057	3.80E-4
V/3A		962′/985′ Recirc Pumps MG Set Room	1.29E5	1.02E-2
VI/10	(Screened)	Admin Bldg Outside Other Fire Compartments	80,000	1.00E-3**
VI/11		965′ Admin HVAC Room	1,303	1.00E-3**
VI/7A	VI/BS7 (Burn Seq. 7)	928′ Admin Battery Room, D-1 & D-5	44,623	2.34E-3
VI/7B		928′ Admin Battery Room, D-3	50,638	1.68E-3
		Total for Sub-area VI/BS7		4.02E-3
VI/8		939′ Admin Cable Spreading Room	59,092	4.40E-3
VII/7C		928′ Admin Battery Room, D-2, D-4	56,289	1.13E-3
VIII/9		951′ Admin Control Room	9,605	9.76E-3
IX/12A		911′ Lower [illegible] Area	25,936	5.35E-3
IX/13A	IX/BS4 (Burn Seq. 4)	911′ TBNE Lube Oil Storage Tank Rm.	9.15E6	3.35E-3*
IX/13B		911′ TB Lube Oil Res & RX Feed Pump	6.48E5	9.02E-3
IX/13C		911′ TBSE ESF Motor Control Center	45,573	4.34E-3
IX/16		911′/931′ Turbine Bldg Corridor	36,155	1.80E-3

Table B.2.6.3 (Continued) Monticello Ignition Source Frequencies and IPEEE Fire Area/Zone Definitions

APPENDIX R FIRE AREA/ZONE	IPEEE FIRE SUB-AREA	ZONE DESCRIPTION	HEAT LOAD (Btu/ft^2)	IGNITION SOURCE FREQ. (per year)
IX/19C		931′ TBSE Pipe & Cable Tray Pen. Area	51,863	1.80E-4
IX/23B		916′ TB/Intake Structure Corridor	6,134	1.50E-4
IX/37		931′ TBA Turbine Bldg Addition	11,208	5.30E-4
			Total for Sub-area IX/BS4	1.60E-2
IX/23A		919′ Intake Structure Pump Room	10,379	1.02E-2
X/22	(Screened)	930″ Recombiner Bldg.	82	9.00E-2
X/12B	X/BS9 (Burn Seq. 9)	911′ TB H2 Seal Oil, Pumps, Cond Pumps	27,496	2.67E-3
X/12C		911′ TB Basement Condenser Area	27,068	4.22E-3
X/12D		908′ TBSE Clean Radwaste Sump Area	728	1.36E-3
X/12E		911′ TB Air Ejector Room	232	3.20E-4
X/14B		931′ TB Valve Operating Gallery	1,059	2.40E-3
X/14C		931′ TBSE Railroad Car Shelter	30,547	3.05E-3
X/30		951′ Turbine Bldg Operating Floor	750	1.21E-2
			Total for Sub-area X\BS9	1.21E-2***
XI/15E	(Screened)	Diesel Fuel Oil Pump House	2,333	Not Calculated
XII/14A		931′ Upper 4KV Area	13,372	5.37E-3
XII/17	XII/BS5 (Burn Seq. 5)	941′ Turbine Bldg Cable Way	43,557	4.40E-4
XII/19A		931′ TB Water Treatment Area	37,123	1.21E-3

Table B.2.6.3 (Continued) Monticello Ignition Source Frequencies and IPEEE Fire Area/Zone Definitions

APPENDIX R FIRE AREA/ZONE	IPEEE FIRE SUB-AREA	ZONE DESCRIPTION	HEAT LOAD (Btu/ft^2)	IGNITION SOURCE FREQ. (per year)
XII/19B		931′ TBSE ESF Motor Control Center	40,705	4.97E-3
			Total for Sub-area XII/BS5	6.62E-3
XII/18A	(Screened)	931′ TBNE Hot Machine Shop	2,977	1.63E-3
XII/18B		931′ TBNE Hot Machine Shop Oil Storage Room	7.10E5	6.00E-5
XII/20		930′ TBSE Auxiliary Boiler Area	54,347	3.58E-3
XII/34		931′ TB Non-1E Electrical Equip Room	26,415	5.36E-3
XII/35		931′ TB Diesel Generator 13 Room	1.11E5	2.92E-2
XII/36		931′ TB Diesel Generator 13 Day Tank Room	2.64E6	1.00E-4
			Total for Sub-area	3.99E-2
XIII/15A		931′ Standby Diesel Generator #12 Rm.	1.29E5	2.92E-2
XIV/15B		931′ Standby Diesel Generator #11 Rm.	1.12E5	2.91E-2
XV/15C		931′ DG Day Tank Room T-45B	1.50E6	1.00E-4
XVI/15D		931′ DG Day Tank Room T-45A	1.50E6	1.00E-4
XVII/25	(Screened)	Discharge Structure Pump Room	7,115	Not Calculated
XVIII/26		Offgas Stack	453	Not Calculated
XVIII/27		Offgas Retention Bldg	1,800	Not Calculated
XIX/28		Guard House	NA	Not Calculated
XX/29		Security Diesel Bldg	1.35E5	Not Calculated

Table B.2.6.3 (Continued) Monticello Ignition Source Frequencies and IPEEE Fire Area/Zone Definitions

APPENDIX R FIRE AREA/ZONE	IPEEE FIRE SUB-AREA	ZONE DESCRIPTION	HEAT LOAD (Btu/ft^2)	IGNITION SOURCE FREQ. (per year)
XXI/31A		EFT 1st Floor, Division I	556	2.90E-4
XXI/32A		EFT 2nd Floor, Division I	4,824	1.99E-3
XXII/31B	XXII/BS6 (Burn Seq. 6)	EFT 1st Floor, Division II	15,973	2.98E-3
XXII/32B		EFT 2nd Floor, Division II	5,494	4.80E-4
		Total for All of Area XXII/BS6		3.46E-3
XXII/33		EFT 3rd Floor Cable Tunnel & Ext Duct	25,595	9.20E-4
XXIII/24	(Screened)	Diesel Fire Pump Room by Intake Structure	81,970	Not Calculated
YARD		Main, 1R, and 2R Transformers	1.05E6	3.00E-3

* L.O. storage tank room not included in IX/BS4 total due to 3-hr barrier and suppression.

** Estimated.

*** Ignition Frequency for quantification of fire area X was limited to the Zone X\30 (1.2E-2) value. This value was used because it contains the majority of cables for safe shutdown equipment and fire spread from the other compartments is unlikely.

B.2.7 Fire Area Initial Screening

A fire in the plant involving equipment that may be required to support plant operation was assumed to result in a plant shutdown. Therefore, only fire areas outside the main reactor/turbine building complex were screened from further evaluation in this step. The results of the qualitative screening process are shown in Table B.2.7.1.

Table B.2.7.1 Summary of Monticello IPEEE Area Screening

IPEEE AREA/ SUB-AREA	DESCRIPTION	QUALITATIVELY SCREENED	RETAINED FOR FURTHER EVALUATION
	Radwaste Building	X	
I/BS8	Rx Bldg Upper Levels		X
I/1B	896′ SE RHR & CS Room - Division I		X
I/2B	935′ CRD Hydraulic Control Area		X
I/2D	935′ SE Rx Bldg Railroad Car Bay	X	
I/2E	935′ NE T.I.P. Room		X
I/2G	935′ East Shutdown Cooling Room		X
I/3B	962′ MCC & Standby Liquid Control Sys Area		X
II/BS1	NW Corner of Rx Bldg basement		X
II/BS2	West side (935′/962′) of Rx Bldg		X
II/1A	896′ SW RHR & CS Room - Division II		X
II/2F	935′ Steam Chase		X
III/BS3	RCIC & T.I.P. Drive Rooms		X
IV/1F	896′ Suppression Pool Area		X
V/3A	962′/985′ Recirc Pumps MG Set Room		X
VI/10 & 11	Admin Bldg Outside Other Fire areas, Control Room or Cable Spreading Room	X	
VI/BS7	928′ Admin Battery Rooms D-1, D-3		X
VI/8	939′ Admin Cable Spreading Room		X
VII/7C	928′ Admin Battery Room, D-2 & D-4		X
VIII/9	951′ Admin Control Room		X

Table B.2.7.1 (Continued) Summary of Monticello IPEEE Fire Screening

IPEEE AREA/ SUB-AREA	DESCRIPTION	QUALITATIVELY SCREENED	RETAINED FOR FURTHER EVALUATION
IX/12A	Lower 4KV Area		X
IX/BS4	Remainder of TB Fire Area IX except Intake Struct Pump Room (23A)		X
IX/23A	919′ Intake Struct Pump Room		X
X/22	Recombiner Building	X	
X/BS9	Turbine Bldg Fire Area X		X
XI/15E	Diesel Fuel Oil Pump House	X	
XII/14A	Upper 4KV Area		X
XII/BS5	TB Fire Area XII (Zones 17, 19A, 19B)		X
	All of Fire Area XII located outside the Turbine Bldg (18A, 18B, 20, 34)	X	
XIII/15A	931′ Standby DG 12 Room		X
XIV/15B	931′ Standby DG 11 Room		X
XVI/15C	931′ DG Day Tank Room T-45B		X*
XVI/15D	931′ DG Day Tank Room T-45A		X*
XVII/25	Discharge Structure Pump Room	X	
XVIII/26&27	Offgas Building	X	
XX/28	Guard House	X	
XX/29	Security Diesel Bldg	X	
XXI/31A& 32A	EFT Bldg. Division I spaces		X
XXII/BS6	EFT Bldg. Division II spaces		X
XXII/33	EFT 3rd Floor, Cable Tunnel and Ext Duct		X

Table B.2.7.1 (Continued) Summary of Monticello IPEEE Fire Screening

IPEEE AREA/ SUB-AREA	DESCRIPTION	QUALITATIVELY SCREENED	RETAINED FOR FURTHER EVALUATION
XXIII/24	Diesel Fire Pump Room by Intake Structure	X	
YARD	Transformers - Main, 1R, and 2R	X**	

* Because of extremely low ignition source frequencies, these rooms were analyzed in combination with their respective Diesel Generator rooms.

** Fire spread between transformers and between transformers and the building is not credible per Appendix R Engineering analyses. Independent failure of transformers are accounted for in the IPE.

B.2.8 Fire Detection and Suppression

This section discusses automatic detection and automatic or manual fire suppression at Monticello. The detection and suppression systems available in each fire area are presented in the Monticello Updated Fire Hazards Analysis [7] and listed in Table B.2.8.1.

While detection and suppression capability are discussed for most areas of the plant, it should be noted that the only locations where detection and/or suppression were credited in the accident sequence quantification were the control room, cable spreading room, and feedwater pump area. It should also be noted that the assumptions and methodology employed in the fire IPEEE, specifically those dealing with suppression of fire in control room panels, are not necessarily the same as those employed in the Appendix R analysis.

B.2.8.1 Detection

Several methods of automatic fire detection are used at Monticello. These methods are ionization detection, heat detection, and flame detection. Alarms are designed to sound locally and in the control room for all zones except for two which alarm in the guard house. The detection system will also sound an alarm if there is a failure in the detector system.

The control room also has numerous battery-operated smoke detectors in cabinets and at the ceiling. These detectors sound alarms locally.

In addition to the alarms described above, there are heat-actuated device alarms and/or water flow alarms associated with water suppression systems which alarm in the control room. These alarms are identified in Table B.2.8.2.

B.2.8.2 Automatic Suppression

The automatic suppression systems at Monticello consist of water and Halon based systems. The water supply for the water suppression portion of the fire protection system consists of a diesel-driven pump and two electric motor-driven pumps that will each deliver 1500 gpm at 90 psi. The water delivery portion of the system consists of automatic pre-action, deluge, wet/dry pipe sprinklers and hose stations.

Although several locations in the plant are protected by automatic fire suppression systems, the cable spreading room and feedwater pump area are the only locations in which this analysis takes credit for the automatic suppression of a fire. The cable spreading room is equipped with a totally flooding Halon fire extinguishing system. This system uses a main bank of four bottles of Halon with a reserve bank of four bottles located in the heating boiler room. The unavailability of the Halon system used in the quantification of this fire scenario is taken from the FIVE methodology. This generic Halon system unavailability is 5E-2.

The feedwater pump area is equipped with a fusible link wet sprinkler fire extinguishing system. This system provides localized coverage over the main feedwater pumps and the TG oil area. The unavailability of the wet sprinkler system used in the quantification of this fire scenario is taken from the FIVE methodology. This generic system unavailability is 2E-2.

B.2.8.3 Manual Suppression

Each nuclear power plant is required to maintain a manual fire fighting capability. The fire brigades developed under these requirements are well trained and capable of fighting fires while awaiting support from professional fire fighting teams, if called. To take credit for brigade or other manually actuated suppression system response in the FIVE methodology, however, the plant must demonstrate that the fire brigade can assemble, fight, and control a fire in the compartment before the fire causes damage to safe shutdown equipment. That is, the time to detect a fire plus the time to respond to the scene with equipment and control the fire must be less than the time required for fire to damage critical equipment.

Detection time is dependent upon the type of detection equipment in a compartment. Ionization detectors should detect a fire during the incipient stages, whereas heat detectors would not be expected to detect a fire until the fire is more fully involved. Fire brigade response time includes time to verify the detection and the time for the team to respond to the scene with equipment. Response time is obviously highly variable and is dependent upon the location of the fire, location of the brigade members at the time of the event and many other factors.

The FIVE methodology assigns a probability of successfully suppressing a fire manually if and only if the following two criteria can be met:

1) The plant can demonstrate that detection and manual response can occur before damage to safe shutdown equipment, and

2) Fire brigade effectiveness can be demonstrated per the requirements of the NUREG/CR-5088 "Fire Risk Scoping Study" [11].

The FIVE methodology states that the probability of manually suppressing a fire should not be greater than 0.9.

For the purpose of this analysis, no credit for manual suppression was taken before damage of safe shutdown equipment is assumed to occur. This analysis recognizes that manual suppression efforts will be taken to suppress a fire and to ensure that the fire does not propagate outside the fire area boundaries. Manual fire suppression equipment is available throughout the plant in the form of portable fire extinguishers and hose stations. The fire fighting training program in place at Monticello ensures that fire brigade members are adequately trained to effectively use this equipment. The limited credit assumed for manual suppression of fires in the Monticello Fire IPEEE is for accident sequence quantification purposes only.

Following successful suppression of a fire in the control room or cable spreading room, some equipment was assumed to be lost. If the fire started in a cabinet or panel, all the circuits in that cabinet/panel were assumed to fail either directly by the fire or indirectly due to suppression-induced damage. Control or cable spreading room fires that did not start inside a cabinet/panel were assumed to fail one system, feedwater (see section B.2.10.3(1)), even if successfully suppressed. Since suppression was not credited except in these limited areas, suppression-induced damage outside of these areas is not an issue.

In the control room, fire detection can be accomplished in a variety of ways:

- The control room cabinets contain local smoke detectors which would provide an audible alarm should smoke be generated within the cabinets.

- The control room contains local smoke detectors in the ceiling which would provide an audible alarm should smoke be generated in the control room.

- The control room is continuously staffed and a fire should be quickly sensed by smell or by sight by the operators.

It is assumed that the failure to detect a fire in the control cabinets is negligibly small due to the redundancy and diversity of cues and due to the continuous staffing of the control room. It is further assumed that fire suppression efforts would be initiated immediately upon detection of a fire because of the continuous staffing of the control room. The FIVE methodology allows a minimum value of 0.1 for the probability of failing to suppress a fire manually in a given space even if unoccupied. This analysis assumes additional credit in the likelihood of successfully suppressing a control room fire for the following reasons:

- The control room is continuously staffed. In addition, control room operators are trained in fire suppression techniques. Therefore, early detection and action to suppress a fire is very likely.

- The cabinets contain relatively small amounts of combustible material.

For these reasons, a probability of 0.01 is assigned to failing to manually suppress a fire in the control room.

Table B.2.8.1 Fire Detection and Suppression

APPENDIX R FIRE AREA/ZONE	ZONE DESCRIPTION	DETECTION	SUPPRESSION
I/1B	896′ SE RHR and CS Room - Division I	Smoke	Ext, Hose (1)
I/2B	935′ CRD Hydraulic Control Area	Smoke	Ext, Hose
I/2D	935′ SE Reactor Bldg Railroad Car Shell	None	Ext (1), Hose (1)
I/2E	935′ NE T.I.P. Room	None	Ext (1), Hose (1)
I/2G	935′ East Shutdown Cooling Room	Smoke	Ext (1), Hose (1)
I/3B	962′ MCC & Standby Liquid Control Sys Area	Smoke	Ext, Hose
I/3E	962′ Contaminated Records Storage	None	Ext (1), Hose (1)
I/4A	985′ Equipment Hatch Area	Smoke	Ext, Hose
I/4B	985′ Cooling Water HX Area	Smoke	Ext, Hose
I/4C	985′ Corridor Outside Main Exh Plenum	Smoke	Ext (1)
I/4D	985′ Standby Gas Treatment Room	Smoke	Ext (1), Hose (1)
I/4E	985′ Reactor Plenum Room	None	Ext, Hose
I/5A	1001′ S Laydown and Decon Area	Smoke	Ext, Hose
I/5B	1001′ N Contaminated Equip Storage Area	Smoke	Ext, Hose
I/5C	985′/1001′ Skimmer Surge Tank Area	Smoke	Ext (1), Hose (1)
I/6	1027′ Refueling Floor	Smoke (partial)	Ext, Hose
I/21A	935′ Radwaste Control Rm.	None	Ext, Hose (1)
I/21B	935′ RWB Baler & Dry Waste Storage	None	Ext, Hose
I/21C	935′ RWB Radwaste Shipping Area	Smoke/Heat	Ext & Hose, Fixed - partial pre-action
I/21D 935′	935′ RWB Low Level Radwaste Storage	None	Ext, Hose (1)
I/21D 947′	947′ RWB Radwaste Bldg at 947′ Elevation	None	Ext, Hose

Table B.2.8.1 (Continued) Fire Detection and Suppression

APPENDIX R FIRE AREA/ZONE	ZONE DESCRIPTION	DETECTION	SUPPRESSION
I/21D 962'	962' RWB SW Vent & RW Tank Area	None	Ext, Hose
I/21D 985'	985' RWB SW RW Area - Fuel Pool Waste Processing	None	Ext, Hose (1)
II/1D	896' NW Equip & Floor Drain Tank Room	None	Ext (1), Hose (1)
II/1E	896' HPCI Room	Smoke	Ext, Hose
II/1G	923' NW CRD Pump Room	None	Ext, Hose (1)
II/2C	935' CRD HCU and HVAC Area	Smoke	Ext, Hose
II/2H	West Shutdown Cooling Room	Smoke	Ext, Hose (1)
II/3C	962' MCC Area	Smoke	Ext, Hose
II/3D	962' Cooling Water Pump & Chiller Area	Smoke	Ext, Hose
II/1A	896' SW RHR & CS Pump Room - Division II	Smoke	Ext, Hose
II/2F	935' Steam Chase	None	Ext, Hose (1)
III/1C	896' NE RCIC Room	Smoke	Ext, Hose (1)
III/2A	935' NE T.I.P. Drive Room	Smoke	Ext, Hose (1)
IV/1F	896' Suppression Pool Area	Smoke	Ext, Hose
V/3A	962'/985' Recirc Pumps MG Set Room	HAD	Ext & Hose. Automatic water spray sys protects MG couplings
VI/10	Admin Bldg Outside Other Fire Compartments	None	Ext, Hose
VI/11	965' Admin HVAC Room	None	Ext, Hose (1)
VI/7A	928' Admin Battery Room, D-1 & D-5	Smoke	Ext (1), Hose (1)
VI/7B	928' Admin Battery Room, D-3	Smoke	Ext (1), Hose (1)

Table B.2.8.1 (Continued) Fire Detection and Suppression

APPENDIX R FIRE AREA/ZONE	ZONE DESCRIPTION	DETECTION	SUPPRESSION
VI/8	939′ Admin Cable spreading room	Smoke/Heat	Ext & Hose (1), Halon flooding sys. Automatic suppression system credited.
VII/7C	928′ Admin Battery Room, D-2, D-4	Smoke	Ext (1), Hose (1)
VIII/9	951′ Admin Control Room	Smoke	Ext, Hose (1) Manual suppression credited.
IX/12A	911′ Lower 4KV Area	Smoke	Ext, Hose
IX/13A	911′ TBNE Lube Oil Storage Tank Room	Sprinkler water flow alarm	Ext (1), Hose (1) Automatic wet pipe sprinkler system
IX/13B	911′ TB Lube Oil Res & RX Feed Pump Area	HADs that trip deluge system	Ext & Hose, Automatic deluge system protects L.O. reservoir
IX/13C	911′ TBSE ESF Motor Control Center	Smoke	Ext, Hose
IX/16	911′/931′ Turbine Bldg Corridor	Smoke	Ext, Hose
IX/19C	931′ TBSE Pipe & Cable Tray Pen. Area	Smoke	Ext (1), Hose (1)
IX/23B	916′ TB/Intake Structure Corridor	None	Ext, Hose (1)
IX/23A	919′ Intake Structure Pump Room	Smoke/Heat	Ext & Hose, Automatic pre-action system
IX/37	931′ TBA Turbine Bldg Addition	None	Ext, Hose
X/12B	911′ TB H2 Seal Oil, Pumps, Cond Pumps	Sprinkler water flow alarm	Ext & Hose, Wet pipe system protects H_2 seal oil unit
X/12C	911′ TB Basement Condenser Area	Sprinkler water flow alarm	Ext & Hose, Wet pipe automatic system

Table B.2.8.1 (Continued) Fire Detection and Suppression

APPENDIX R FIRE AREA/ZONE	ZONE DESCRIPTION	DETECTION	SUPPRESSION
X/12D	908' TBSE Clean Radwaste Sump Area	None	Ext, Hose (1)
X/12E	911' TB Air Ejector Room	None	Ext (1)
X/14B	931' TB Valve Operating Gallery	None	Ext, Hose (1)
X/14C	931' TBSE Railroad Car Shelter	None	Ext, Hose
X/22	930' Recombiner Bldg	None	Ext, Hose (1)
X/30	951' Turbine Bldg Operations Floor	None	Ext, Hose
XI/15E	Diesel Fuel Oil Pump House	None	Ext
XII/14A	931' Lower 4KV Area	Smoke	Ext, Hose
XII/17	941' Turbine Bldg Cable Way	Smoke	Ext (1), Hose (1)
XII/18A	931' TBNE Hot Machine Shop	None	Ext, Hose
XII/18B	931' TBNE Hot Machine Shop Oil Storage Room	Sprinkler water flow alarm	Ext (1), Hose (1), Automatic wet pipe system
XII/19A	931' TB Water Treatment Area	Smoke	Ext, Hose
XII/19B	931' TBSE ESF Motor Control Center	Smoke	Ext, Hose
XII/20	930' TBSE Auxiliary Boiler Area	Heat	Ext, Hose
XII/34	931' TB Non-1E Electrical Equip Room	Smoke	Ext, Hose (1)
XII/35	931' TB Diesel Generator 13 Room	Heat/Flame	Ext & Hose (1), Automatic dry pipe pre-action system
XII/36	931' TB Diesel Generator 13 Day Tank Room	Heat/Flame	Ext (1), Hose (1) Automatic dry pipe pre-action system
XIII/15A	931' Standby Diesel Generator #12 Room	Heat/Flame	Ext (1), Hose (1), Pre-action sprinkler system

Table B.2.8.1 (Continued) Fire Detection and Suppression

APPENDIX R FIRE AREA/ZONE	ZONE DESCRIPTION	DETECTION	SUPPRESSION
XIV/15B	931' Standby Diesel Generator #11 Room	Heat/Flame	Ext & Hose, Pre-action sprinkler system
XV/15C	931' DG Day Tank Room T-45B	Heat/Flame	Ext (1), Hose (1), Pre-action sprinkler system
XVI/15D	931' DG Day Tank Room T-45A	Heat/Flame	Ext (1), Hose (1), Pre-action sprinkler system
XVII/25	Discharge Structure Pump Room	None	Ext
XVIII/26	Offgas Stack	None	Ext
XVIII/27	Offgas Retention Bldg	None	Ext
XIX/28	Guard House	Heat/Smoke	Ext & Hose, Wet pipe system and halon system in computer room
XX/29	Security Diesel Bldg	Heat/Smoke	Ext, Pre-action system protects diesel and oil tank room
XXI/31A	EFT 1st Floor, Division I	Smoke	Ext, Hose
XXI/32A	EFT 2nd Floor, Division I	Smoke	Ext, Hose
XXII/31B	EFT 1st Floor, Division II	Smoke	Ext, Hose
XXII/32B	EFT 2nd Floor, Division II	Smoke	Ext, Hose
XXII/33	EFT 3rd Floor Cable Tunnel & Ext Duct	Smoke	Ext, Hose
XXIII/24	Diesel Fire Pump Room by Intake Structure	None	Ext
YARD	Transformer Main, 1R, and 2R Transformers	Heat	Ext & Hose (1), Deluge water sys over xfmers and TB exterior wall

Table B.2.8.2 Heat Activated Devices/Water Flow Alarms in Main Control Room

LOCATION	ALARM
Main Transformer (YARD)	HAD
Auxiliary Transformer (YARD)	HAD
Reserve Transformer (YARD)	HAD
Turbine Building Siding (YARD)	HAD Deluge Water Flow
Cooling Towers	HAD Deluge Water Flow
Lube Oil Drum Storage (18B)	Sprinkler Water Flow
Lube Oil Storage Tanks (13A)	Sprinkler Water Flow
Hydrogen Seal Oil (12B)	Sprinkler Water Flow
Lube Oil Piping Under Turbine (12C)	Sprinkler Water Flow
Recirculation MG Set (3A)	HAD Deluge Water Flow
Lube Oil Reservoir (13B)	Deluge Water Flow
Diesel Generator Building (15A)	Pre-action Water Flow
Diesel Generator Building (15B)	Pre-action Water Flow
Diesel Generator Day Tank Room (15C)	Pre-action Water Flow
Diesel Generator Day Tank Room (15D)	Pre-action Water Flow
Intake Structure (23A)	Pre-action Water Flow
Diesel Generator #13 Enclosure (35)	Pre-action Water Flow

B.2.9 Fire Growth and Propagation

All potential propagation paths that could result in fire spreading to a compartment containing safe shutdown equipment or plant trip initiators were considered. The Appendix R fire areas and zones were reviewed to assess the potential for cross area/zone propagation based on the existing fire barriers and fire zone loading.

The potential for fire spread from the compartment being evaluated (exposing compartment) to the adjacent compartments (exposed compartments) was then examined. Each common boundary was analyzed for fire spread in either direction. A means of addressing fire spread across these boundaries is addressed in the FIVE methodology and was used in this study. Criteria to determine fire spread were identified in Section B.2.1.

Any scenario where a fire could potentially involve two or more adjacent zones was analyzed for potential fire spread by extending unscreened boundaries. This step was performed in accordance with the FIVE screening criteria shown in Section B.2.1. Fire spread scenarios were identified and tracked for all entered fire zones.

Fire scenarios that have the potential to spread beyond the initiating compartment were identified as Burn Sequences in Table B.2.6.3. There are nine locations within the plant that have the potential for fire spread beyond the originating compartment. Fires in compartments not shown in this table will not spread to adjoining compartments.

B.2.10 Fire Event Trees

This analysis was based upon the Monticello transient event tree from the internal events PRA (Figure B.2.10-1). A fire in most locations in the Monticello plant would initiate an event similar to a transient event with one or more of the systems identified in Section B.2.1 out of service due to the fire. One additional event tree was developed specifically for this analysis (Figure B.2.10-2). It was developed for fires in the main control and cable spreading rooms and was based on the internal events PRA transient event tree. Top events were added to account for the effects of suppression and switching control of the plant to the alternate shutdown system panel.

Accident classes were defined such that core damage sequences with similar characteristics (e.g., reactor vessel failure pressure, core damage timing, system failures) could be grouped and analyzed together. The three accident classes employed in the fire IPEEE are a subset of the accident classes found in the internal events PRA. These accident classes are Class 1A - loss of coolant makeup with core melt at high pressure; Class 1D - loss of coolant makeup with core melt at low pressure; and Class 2 - loss of decay heat removal.

B.2.10.1 Fire Event Tree Top Event Definitions

FIRE Fire Initiator

The fire is defined as starting in a location that would cause a plant transient initiator, require a manual shutdown, or affect plant equipment potentially useful for plant shutdown.

C Reactivity Control

All 121 control rods fully insert into the reactor core.

M SRV Open

Sufficient safety/relief valves open to relieve the pressure transient. Fire events require at least one of the eight SRVs to open.

P SRV Close

All SRVs that opened to relieve the pressure transient close when pressure is reduced below the normal valve closure setpoint.

SUP Suppression of Fire Before Spread (control/cable spreading rooms only)

The fire is suppressed by either occupants in the room or by automatic suppression equipment before it can spread to other locations. In the control room fire, successful manual suppression limits the extent of the fire to the cabinet in which it is assumed to initiate. Successful suppression of a fire in the cable spreading room assumes that fire damage occurs to at least one system (feedwater) but is limited to that system. It is assumed that the smoke created from a fire involving one panel, given the existing ventilation, would not force the evacuation of the control room. In the event of fire suppression failure in the control room or cable spreading room, it is assumed that extensive damage is possible, requiring reactor depressurization and inventory makeup to be accomplished from the ASDS panel. Controls for Division II SRVs, RHR B and core spray B are provided on the ASDS panel.

QU High Pressure Coolant Injection

Injection of coolant at high pressure from continued operation of feedwater, from HPCI, or from RCIC.

ASDS Operators Control Plant at ASDS Panel (control/cable spreading rooms only)

The operators carry out the "Shutdown Outside Control Room" procedure, Operations Manual C.4-C, evacuating the main control room (immediate actions) and transferring plant control to the ASDS panel (supplementary actions, steps 1-5).

X Reactor Depressurization

Subsequent to failure of high pressure injection systems, the reactor must be depressurized to allow low pressure injection systems to function. This is accomplished by actuation of at least one of the eight SRVs.

V Low Pressure Coolant Injection

Subsequent to failure of high pressure injection and successful reactor depressurization, the low pressure injection systems function as in any other transient scenario with LPCI, core spray, the fire protection system, or condensate providing makeup.

CRD Control Rod Drive Pumps

CRD pumps are assumed to supply adequate makeup only following successful injection by other higher volume sources. They are only credited for late injection requirements (i.e., Class 2) because of their relatively low capacity.

W Decay Heat Removal
Heat removal is normally accomplished by the main condenser, suppression pool cooling, or shutdown cooling. The hard pipe vent can be used as a means of containment pressure control should all other decay heat removal systems become ineffective.

QUV Coolant Injection After Containment Failure
Following containment failure, feedwater, CRD or HPCI could potentially provide coolant makeup. Low pressure systems are not credited because containment pressure is assumed to remain above the pressure at which SRVs will remain open (70 psig).

B.2.10.2 Event Tree For Fire in Main Control Room

The event tree for fire in the main control room is similar to the transient event tree. The differences are:

(1) The event SUP is included to account for the likelihood of the fire being suppressed by the operators before it can spread from a single control room panel. This event was discussed in Section B.2.8. The probability assigned to the failure of this event is 0.01.

(2) The event ASDS is included to account for the operators' ability to recognize the need to evacuate the control room and to successfully transfer control of the plant to the ASDS panel and control the plant from the ASDS panel. A human reliability analysis was performed on this action. The probability of failure of this event is 3.4E-3, given at least thirty minutes to staff the ASDS panel.

This event tree assumes that suppression of the fire in the control room must be unsuccessful before it can spread to other locations. Successful manual suppression, therefore, limits the extent of the fire to the cabinet in which it initiates or to localized damage if it starts outside of a cabinet. Spreading of the fire beyond the initiating cabinet is assumed to force the evacuation of the control room.

This event tree was quantified using the same methods used in the internal events PRA for the transient tree and the results of that quantification are provided in Table B.2.11.1.

B.2.10.3 Event Tree For Fire in Cable Spreading Room

The event tree for fire in the cable spreading room is also similar to the transient event tree. The differences are:

(1) The event SUP is included to account for the likelihood of the fire being suppressed before it can spread to locations impacting more than one injection system. Automatic suppression is assumed to limit the extent of the fire to the cabling of a single system, feedwater. The feedwater system was selected to be failed during cable spreading room fires that were suppressed because its loss has the highest impact on the core damage frequency of any of the injection systems credited in the analysis of this area. Feedwater also causes a plant trip if it fails. The probability assigned to the failure of cable spreading room suppression is estimated to be 5E-2 as discussed in Section B.2.8.

(2) The event ASDS is included to account for the operators' ability to recognize the need to successfully transfer control of the plant to the ASDS panel and to shut down the plant from that location. This is the same event as described above for the control room fire and the probability of failure of this event is estimated to be 3.4E-3.

It is assumed in this analysis that when the fire is suppressed by automatic suppression equipment, it does not spread to other locations. Automatic suppression, therefore, limits the extent of the fire to the feedwater cabling.

This event tree was quantified using the same methods used in the internal events PRA for the transient tree and the results of that quantification are provided in Table B.2.11.1.

B.2.10.4 Accident Sequence Classification

This section discusses the classification of core damage sequences into functional categories based upon characteristics of the accident sequences with respect to reactor and containment conditions at the time core damage is assumed to occur. These functional categories are called "accident classes".

The potential types and frequencies of accident scenarios at a nuclear power plant cover a broad spectrum. In order to limit these sequences to a manageable number, sequences with similar functional characteristics are grouped together. Three such functional classes were defined for the Monticello fire IPEEE:

Class 1A - Transient-initiated events in which all high pressure injection systems become unavailable and depressurization of the reactor to allow low pressure injection is not accomplished. Core damage is assumed to occur with the reactor at high pressure for these sequences.

Class 1D - Transient-initiated events in which all high and low pressure injection systems become unavailable. Depressurization of the reactor is successful for these sequences. Core damage is assumed to occur at a low reactor pressure.

Class 2 - Core damage events which occur as a result of the inability to remove decay heat from the containment. All means of heat removal are assumed not to function for this accident class, including the main condenser, containment venting, and RHR in shutdown cooling, suppression pool cooling, and wetwell and drywell spray modes. This accident sequence takes days to develop, saturation of the pool taking more than eight hours, pressurization to containment design on the order of a day, and closure of SRVs prohibiting low pressure injection at least thirty hours into the event. High pressure systems must also fail to result in core damage for this accident class.

These accident classes are typical of other PRAs and are a subset of those used in the Monticello internal events PRA. Other accident classes that were not considered to be applicable to the fire PRA include:

Class 1B - Station blackout. No single fire area is likely to result in a loss of all AC power at Monticello.

Class 3 - LOCAs. No fire initiator was identified that could credibly lead to a loss of coolant accident.

Class 4 - ATWS. No fire initiator was identified that could credibly lead to a failure of the reactor protection system. The simultaneous, independent failure of the reactor protection system or of control rod insertion during a fire is probabilistically insignificant.

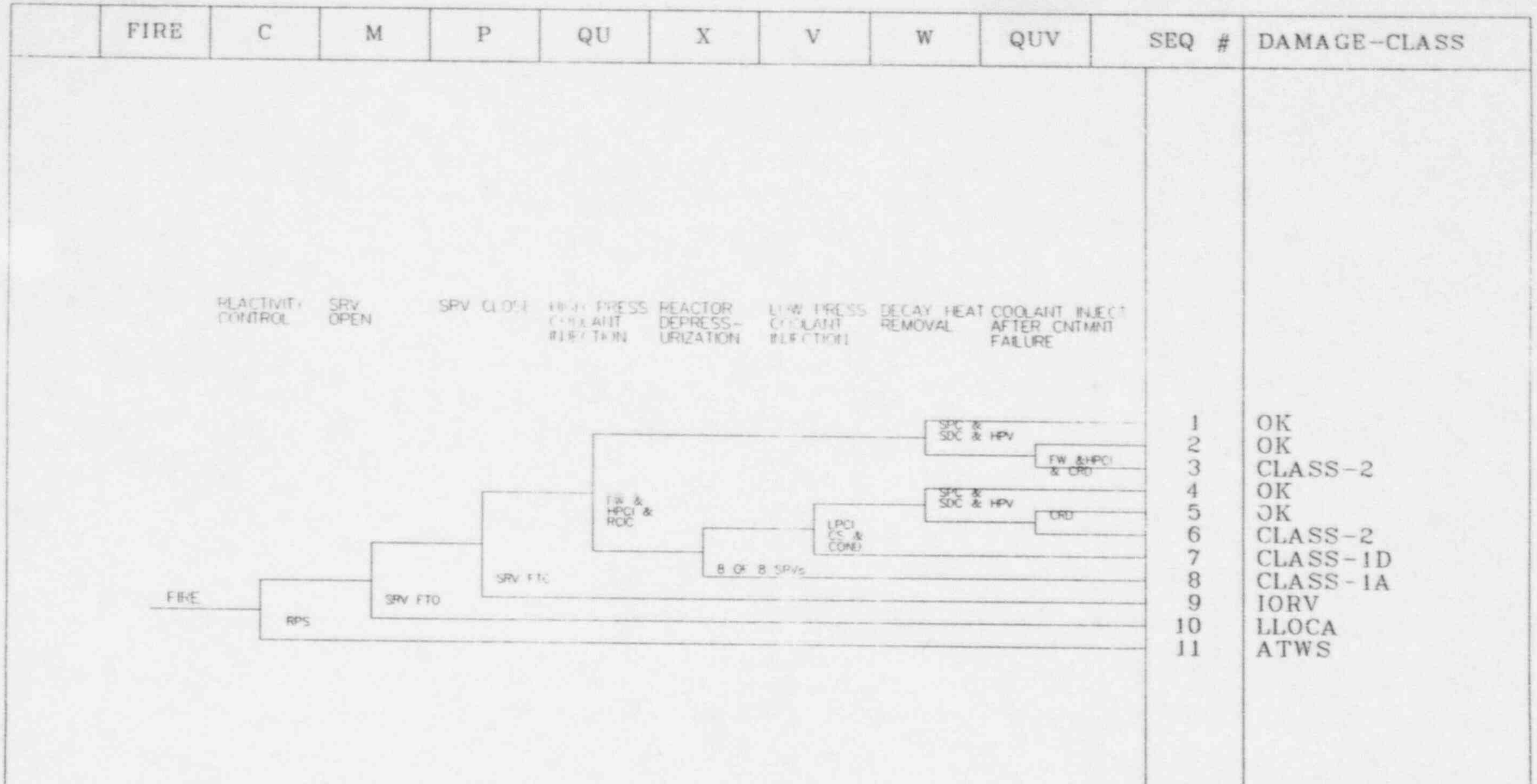


Figure 2.10-1
Transient Event Tree

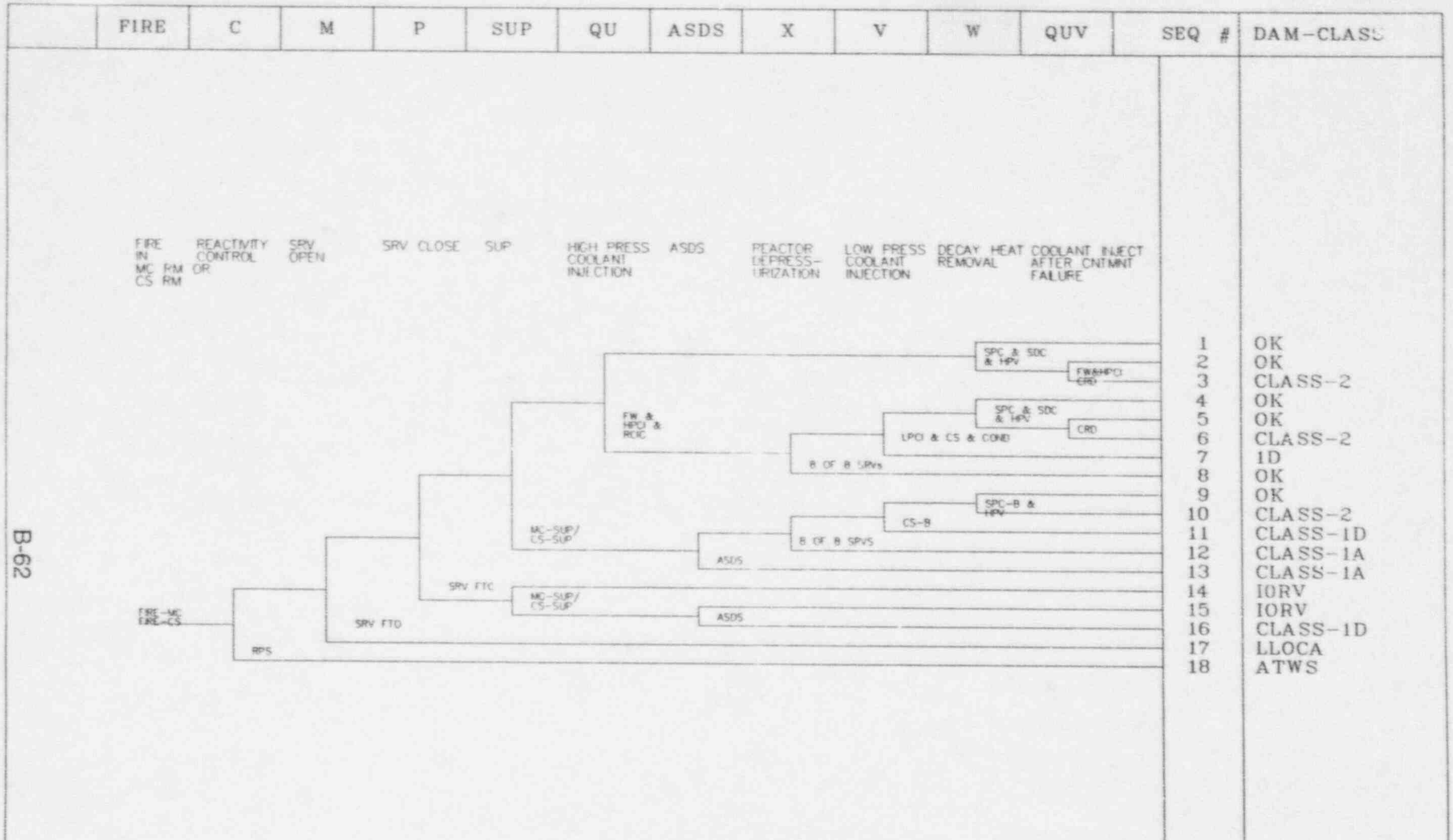


Figure 2.10-2
Fire Event Tree
Control Room and Cable Spreading Area

B.2.11 Analysis of Fire Sequences and Plant Response

The following screening criteria were used to identify sequences to be discussed in this section of the report. These criteria are identical to the functional reporting requirements presented in Generic Letter 88-20 as required by NUREG-1407.

(1) Functional sequences with a CDF greater than 1E-6 per year. (Functional sequences for the Monticello fire IPEEE are the three accident classes defined in Section B.2.10).
(2) Functional sequences that contribute five percent or more to total CDF.
(3) Sequences determined by the utility to be important contributors to CDF or containment performance.

Although these reporting criteria are suggested by NUREG-1407, all three functional accident classes quantified in the Monticello fire IPEEE are described regardless of whether they meet the screening thresholds.

B.2.11.1 Important Accident Classes

Class 1A: The sequences within this class were characterized by a loss of high pressure inventory makeup with a failure to depressurize the reactor vessel. Class 1A sequences had a total CDF of 2.9E-6/year, or 37% of the overall internal fire events CDF. The Class 1A sequences were dominated by sequences initiated by fires in the turbine building 931' area (fire zones XII/17, 19A and 19B), fires in the control/cable spreading rooms, MCC 133/feedwater pump area (fire zones IX/13B, 13C and 19C) and in the switchgear rooms.

For these sequences, reactivity control is successful and the SRVs cycle to control primary system pressure. Steam flow to the suppression pool through the relief valves is assumed to occur periodically throughout the event because of decay heat. High pressure injection would be the first function lost for these sequences. If high pressure injection fails, reactor vessel depressurization is required to permit low pressure injection. Without reactor vessel depressurization, the top of the fuel is assumed to be uncovered about 25 to 35 minutes after the initiating event.

Important assumptions applicable to this class that are reiterated from the internal events PRA are:

1. The operator always inhibits the automatic depressurization system (ADS) as directed by the emergency operating procedures. Inhibiting ADS, making depressurization a manually controlled action, was considered to be a conservative and bounding assumption.

2. No credit was taken for recovering HPCI and RCIC before core damage occurred.

3. The end state for this accident class is reactor water level at the top of active fuel and the initiation of core damage with the reactor at high pressure. These conditions are assumed to occur approximately 25 minutes after loss of injection systems.

4. A loss of a single train of 125 VDC power was assumed to result in a reactor trip. In reality, loss of DC power would not lead to a reactor trip, but instead to a manual shutdown after several hours, providing time for manual operation of DC-operated components currently not credited.

The most significant fire initiating events were:

1. Fire in the turbine building 931' area (fire zones XII/17, 19A and 19B) accounts for 22% of the contribution to Class 1A CDF. A fire in this area is assumed to fail HPCI as well as one train of feedwater and CRD. Control to condenser hotwell makeup valves also fails, requiring manual alignment of condensate to the hotwell.
   In addition to loss of Division II cables due to the fire, other failures are required before core damage would occur. In this accident class, a concurrent random failure of power to the other division (e.g., spurious opening of the Division I load breakers from Bus 15 to LC 103, Bus 15 failure, etc.) results in the majority of the core damage frequency associated with this event.

2. Fire in the control/cable spreading rooms accounts for about 24% of the fire-initiated contribution to Class 1A CDF. Failure to suppress the fire (operator inability to manually suppress a fire in the control room and automatic suppression system failure in the cable spreading room) followed by failure to take control of the plant at the ASDS panel dominated these fire scenarios. This combination of events is assumed to result in the unavailability of all reactor makeup and heat removal systems.

3. Fire in the MCC 133/feedwater pump area (area IX, zones 13B/13C/19C) accounts for about 14% of the fire-initiated contribution to Class 1A CDF. A fire in this area is assumed to result in loss of feedwater and a single train of CRD. The only high pressure coolant injection systems unaffected by the fire are HPCI, RCIC and a train of CRD. A number of SRVs may also be disabled by this fire, thus affecting the ability to perform the depressurization function.

   In addition to loss of Division I cables due to the fire, other failures are required before core damage would occur. In this accident class, a concurrent random failure of power to the other division (e.g., spurious opening of the Division II load breakers from Bus 16 to LC 104, Bus 16 failure, etc.) results in the majority of the core damage frequency associated with this event.

4. A fire in the lower 4KV switchgear (zone 12A) has an impact on the plant very similar to a fire in the feedwater pump area and accounts for 13% of the core damage contribution for this accident class. It results in loss of power to Division I equipment.

The most significant operator actions contributing to this accident class were:

1. Failure to repair/reclose key circuit breakers. Fires in turbine building areas that house entire divisional cable runs dominate this accident class. Spurious opening or failure of key breakers in the undamaged division lead to failure of all high pressure

injection sources. Approximately 32% of the damage in this accident class results from failure of the operator to repair/reclose these breakers prior to depletion of the station batteries.

2. Failure to blow down the reactor vessel accounts for 19% of the Class 1A CDF. This relatively low contribution is attributed to the high likelihood that the operator will successfully initiate emergency depressurization when required.

3. Failure to suppress the fire in the control room. Approximately 15% of the core damage associated with this accident class results from sequences in which manual fire suppression in the control room is unsuccessful.

Important hardware failures associated with this accident class include:

1. Division II feeder breakers from Bus 16 to LC 104 (152-609, 52-401). These breakers are normally closed. A spurious opening of feeder breakers to LC 104 concurrent with a fire in area IX (which disables Division I power, LC 103) is assumed to result in loss of both trains of essential 480 VAC supply. This failure combination would result in loss of all safeguard trains that require motive and control power from LC 103 and 104; thus the importance of the feeder breakers. Other important hardware failures include LC 104/Bus 16 faults and failure of transformer X-40 in the Division II AC system.

2. Division I feeder breakers from Bus 15 to LC 103 (152-509, 52-301). These breakers are normally closed. A spurious opening of feeder breakers to LC 103 concurrent with a fire in area XII (which disables Division II power, LC 104) is assumed to result in loss of both trains of essential 480 VAC supply. This failure combination would result in loss of all safeguard trains that require motive and control power from LC 104 and 103; thus the importance of the feeder breakers. Other important hardware failures include LC 103/Bus 15 faults and failure of transformer X-30 in the Division I AC system.

Class 1D: Sequences in this class were characterized by events with successful depressurization but a loss of both high and low pressure inventory makeup systems. Class 1D sequences made up 42% of the fire-initiated CDF at Monticello. They had a combined sequence frequency of 3.2E-6 per year.

Assumptions associated with the Class 1D sequences in general were:

1. Credit was not taken for use of the fire system or RHR service water system for vessel makeup. These systems are manually aligned from outside the control room and inject through the LPCI injection lines.

2. Depressurization could occur at about 10 minutes and the core could become uncovered sooner than in Class 1A sequences because of the action to depressurize the reactor vessel. Core damage was assumed to occur at 25 minutes.

3. The condenser hotwell normally contains 43,000 gallons of water. This water inventory will allow approximately 6-hrs of makeup at decay heat rates following a plant trip. Long term injection requires makeup from the condensate storage tank or service water.

Significant fire initiating events for this accident class were:

1. Unsuppressed fires in the control and cable spreading rooms account for 49% of the fire-initiated contribution to Class 1D CDF. These fires are significant in that Core Spray loop B is the only low pressure injection system controllable from the ASDS panel following control/cable spreading room evacuation.

2. Fire in the turbine building 931' area (fire zones XII/17, 19A and 19B) accounts for 14% of the contribution to Class 1D CDF. A fire in this area is assumed to fail HPCI in addition to one train each of feedwater and CS. LPCI also fails because one end of the LPCI swing bus (MCC 43) is located in this area.

3. Fire in the MCC 133/feedwater pump area (area IX, zones 13B/13C/19C) accounts for about 19% of the fire-initiated contribution to Class 1D CDF. A fire in this area is assumed to result in a complete loss of feedwater and as well as a single train of CS. LPCI also fails because one end of the LPCI swing bus (MCC 33, normal power feed) is located in this area.

The most significant operator actions contributing to this accident class were:

1. Failure to suppress fires in the control room accounts for 30% of the Class 1D CDF contribution. The impact of all other operator actions are small when compared to this action.

Important hardware failures associated with this accident class include:

1. Following unsuppressed control and cable spreading room fires, loop B of Core Spray is the only injection system controllable from the ASDS panel and assumed to be available. Failure of this single system leads directly to a core damage sequence and subsequently accounts for two-thirds of the core damage contribution for this accident class.

Class 2: Class 2 events were accident sequences resulting from a loss of containment heat removal. The core damage probability for this accident class was determined to be 1.65E-6 per year due to fires, or approximately 21% of the total.

Assumptions for Class 2 events were:

1. The Monticello IPEEE assumed that the SRVs would close because of high pneumatic backpressure when drywell pressure rises above 70 psig, requiring the use of high pressure injection systems such as feedwater or CRD for the duration of the event.

2. Containment failure pressure was estimated to be 103 psig. It is recognized that the actual containment failure pressure and failure location are uncertain. For this analysis, the failure is not assumed to be large enough to depressurize containment. This, combined with assumption #1, results in the assumption that low pressure systems such as RHR and core spray are not capable of providing long term makeup. Best-estimate analysis of the structural capability of the containment suggests that the failure location would most likely be in the drywell head area or torus expansion bellows.

3. Even though containment failure is not expected for several days during a total loss of decay heat removal, recovery actions when the pressure is above 56 psig in containment were not credited for personnel safety reasons.

4. The condensate storage tanks normally contain over 220,000 gallons of water and thus can make up for nearly three days of decay heat. This, combined with the capability to make up from numerous other sources (demineralized water, radwaste, condensate makeup, service water), limits the importance of condensate storage tank capacity as a potential failure mode.

5. The success of venting, or the use of the containment sprays with water from sources external to the containment, was assumed to have no negative effect upon the net positive suction head (NPSH) for injection systems taking suction from the suppression pool. The emergency operating procedures instruct monitoring NPSH to protect operation of injection systems. As injection to the reactor is required only at decay heat makeup rates, the potential for low NPSH is further limited.

6. Given the significant time available to initiate decay heat removal (one to two days), restoration or repair was credited for those systems out for maintenance or experiencing random failure. A non-recovery probability was applied based on the equation $e^{-T/R}$ where **T** is time available to perform recovery and **R** is the mean time to repair [9]. A recovery failure probability of 1E-3 was the smallest value used. In instances where the calculated value was less than this amount, a value of 1E-3 was substituted. No credit was taken for repair of equipment damaged by fire.

7. The condenser hotwells normally contain 43,000 gallons of water. This water inventory will allow approximately 6-hrs injection at decay heat rates following a plant trip. Long term injection requires makeup from the condensate storage tank or service water.

Significant fire initiating events for this class were:

1. Fire on the 935/962' elevations of the reactor building (BS2/zones 2C, 2H, 3C, and 3D) accounts for 32% of the damage for this class. Cables necessary for the operation of HPCI, the hard piped vent, shutdown cooling, and Division II of suppression pool cooling are located in this area. In addition, cables associated with both trains of ECCS automatic start signals are located in this area. The many

electrical and mechanical components in this area also results in a high ignition frequency.

2. Fire in the EFT building (zone 31B and 32B) leads to loss of most Division II equipment including the hard pipe vent, HPCI, and one train each of feedwater, shutdown cooling, and suppression pool cooling. Feedwater and suppression pool cooling remain to remove decay heat. Fires in this area account for approximately 17% of the core damage associated with accident Class 2.

3. Fire in the turbine building 931' area (fire zones XII/17, 19A and 19B) accounts for 11% of the contribution to Class 2 CDF. A fire in this area is assumed to fail HPCI as well as one train of feedwater, CRD, and RHR. In addition, power required for operation of the hard pipe vent is failed. Control to condenser hotwell makeup valves also fails, requiring manual alignment of condensate to the hotwell.

4. Fires in the Standby Liquid Control Pump area (fire zone 3B, RB 962') lead to failures of Division I low pressure systems and MCC-11. Random failures of power supply Y30/Y80 or Division II circuit breakers in conjunction with feedwater failures lead to loss of decay heat removal following fires in this area. These failures account for approximately 9% of the contribution for this accident class.

The most significant operator actions in Class 2 were:

1. Failure to manually align condensate to the main condenser which accounts for 23% of the CDF contribution for this accident class. Alignment of condensate to the condenser may consist of opening the SW cross-tie, the manual bypass (FW-44), or normal CST makeup valves (CV-1094).

2. Failure to recover RHRSW from corrective maintenance, which was found in cutsets accounting for 11% of the core damage frequency attributable to this accident class. The RHRSW system is a heat sink for the RHR heat exchangers. Only one train of the system is available in most of the significant fire areas. Loss of the remaining train results in failure of the RHR system to remove decay heat from the primary system or containment. Because of the long time available prior to core damage under loss of decay heat removal conditions (about two days), the potential for repair and recovery was considered where possible for RHRSW components.

Important hardware failures associated with this accident class include:

1. Various failures of the feedwater system account for 34% of the Class 2 contribution. Feedwater pump failures make up 15% of the total with failure of valves SW-145/147 and FW-67-1 contributing 10% and 9% respectively.

2. Unavailability of RHRSW loop 1 due to corrective maintenance was found in cutsets accounting for 14% of the core damage frequency attributable to this accident class. The RHRSW system is a heat sink for the RHR heat exchangers. Only one train of

the system is available in most of the significant fire areas. Loss of the remaining train results in failure of the RHR system to remove decay heat from the primary system or containment. Because of the long time available prior to core damage under loss of decay heat removal conditions (about two days), repair or recovery of these components prior to core damage is very likely.

B.2.11.2 Important Fire Areas/Rooms

Eighty three percent of the plant risk associated with internal fires can be traced to seven rooms/burn sequences. These rooms/burn areas consist of (1) the main control room, (2) turbine building 931'(fire zones XII/17, 19A and 19B), (3) the MCC 133/feedwater pump area, (4) the cable spreading room, (5) the reactor building 935/962' west, (6) the lower 4KV switchgear room and, (7) the Division II area of the EFT building. Table B.2.11.1 provides a detailed breakdown of core damage frequency for internal fires by fire area and accident class.

This section provides the detailed plant response for each fire area/sub-area not previously screened from consideration. The quantification results presented in Table B.2.11.1 include:

(1) The sub-area in which the fire occurs,
(2) The frequency of fire ignition in that area,
(3) The systems/subsystems potentially affected by a fire in that area,
(4) The core damage frequency (CDF) for this fire, assuming all the systems in this specific area have failed,
(5) Amplifying remarks where appropriate.

Control room/cable spreading room (sub-areas VIII/9 & VI/8): The control and cable spreading rooms contain controls, monitoring instrumentation, and cables for most of the equipment used to achieve safe shutdown of the plant. Loss of these areas due to a fire was assumed to disable all equipment that could not be controlled locally or from the ASDS panel.

Most control and cable spreading room fires start in electrical cabinets or panels. Fire damage or subsequent suppression induced damage were assumed to render all circuits within the cabinet inoperable. Fires within enclosed cabinets were assumed not to spread beyond the initiating cabinet, however, it was assumed that the smoke created from a fire that was not successfully suppressed forced the evacuation of the control room.

If the fire was suppressed, all equipment not controlled from that panel was assumed to be available for use and would fail only due to random causes. If the fire was not suppressed, evacuation of the control and cable spreading rooms is assumed necessary and only equipment controlled from the ASDS panel was considered available.

General area fires (i.e., fires initiating outside of enclosed electrical cabinets) within the control and cable spreading rooms were assumed to engulf the entire room if not suppressed. Manual suppression was credited in the control room and automatic suppression was credited in the cable spreading room. If suppression was successful, the cabling associated with at least one system (feedwater) was assumed to be damaged. Suppression therefore limits the extent of fire damage to a single system.

Class 1A and Class 1D sequences comprise the majority of the risk associated with a fire in both the control room and cable spreading room. Class 1A sequences were dominated by operator inability to take co:.trol at the ASDS panel in time to provide adequate core cooling following failure to suppress the fire in the control room or the cable spreading room. This procedure is detailed in NSP Operations Manual procedure C.4-C, "Shutdown Outside Control Room." Core damage from Class 1D sequences required random failure of Core Spray loop B following failures to suppress the fire. Efforts to repair and recover these components were not credited in these accident sequences given the short time frame before core damage would occur.

Turbine building 931'(Sub-area XII/BS5): This sub-area contains the Division II cable runs, the #14 air compressor and several MCCs.

Class 1A and 1D sequences dominate plant risk due to a fire in this area. Cables for Division II of Feedwater, CRD, Core Spray, and RHR are located in adjacent cable trays and would be susceptible to fire damage. In addition, HPCI cables are located in this area. Because most power cables from the Division II load centers (LC102/104) are located in this area, continued operation of the Division I essential load center is important to the safe shutdown of the plant. Concurrent random failures leading to loss of LC103 are therefore required before core damage would occur. Spurious opening of the breakers feeding LC103 (152-509/052-301) and MCC 133 (052-304), in conjunction with the failures caused by the fire, comprise the majority of the plant risk from fires in this area. Because several hours of RCIC injection may be available (until battery depletion), restoration of Division I power was credited.

Feedwater pump area (Sub-area IX/BS4): This sub-area contains the feedwater pumps, two service/instrument air compressors, and several MCCs. In addition, several large cable trays containing Division I power and control cables run the length of this area.

Class 1A and 1D sequences contribute the majority of the plant risk due to a fire in this area. Both feedwater pumps are located in this space and could suffer damage if a fire were to occur. Cables for one train of Core Spray, RHR and CRD are located in adjacent cable trays and would be susceptible to fire damage. Because most power cables from the Division I load centers (LC101/103) are located in this area as well, continued operation of the Division II essential load center is important to the safe shutdown of the plant. Concurrent random failures leading to loss of LC104 are therefore required before core damage would occur. Spurious opening of the breakers feeding LC104 (152-609/052-401), in conjunction with the failures caused by the fire, comprise the majority of the plant risk from fires in this area. Several hours of HPCI and RCIC injection may be available (until battery depletion). Recovery factors to restore battery chargers or Division II power were applied.

Reactor Building 935/962' West (BS2): A fire in this area has the potential to disable several important systems. Division II of RHR, SPC and, CS as well as HPCI, HPV and SDC are failed by fires in this area. Both trains of ECCS automatic start circuitry are also located in this area. Because of the significant quantity of electrical and mechanical equipment located in this area, the ignition frequency is also large. Feedwater, RCIC and Division I low pressure systems are available for injection following fires in this area.

Class 2 sequences contribute the majority of the plant risk due to a fire in this area. Feedwater and RHR Division I are the only systems available to accommodate decay heat generation following a fire. Failures of the operator to control feedwater and unavailabilities of RHR due to maintenance are significant contributors to core damage.

Lower 4KV Switchgear Room: A fire is this area will fail most Division I switchgear and consequently all Division I safety related equipment.

Class 1A and 1D sequences contribute the majority of the plant risk due to a fire in this area. Cables or the power supply for RCIC and one train of Feedwater, Core Spray, RHR and CRD are located in this room and would be susceptible to fire damage. Because most power cables from the Division I load centers (LC101/103) are located in this area, fires in this area are almost identical to fires occurring in the feedwater pump area (see explanation contained in description of feedwater pump area fires).

Division II Area of the EFT Building: This area contains cabling for the HPCI battery, HPV, MCC-44 and, Division II low pressure systems. 125VDC panel #211 cabling is also located in this area and provides breaker control power for most Division II equipment. Systems available for injection include RCIC and one train each of Feedwater, LPCI and Core Spray (CRD was not credited). Many of the Division II cables are located in this area because of routing to the ASDS panel located in an adjoining room.

Class 2 sequences contribute the majority of the plant risk due to a fire in this area. Feedwater and RHR Division I are the only systems available to mitigate decay heat generation following this fire. Similar to burn sequence 2 (BS2), failures of feedwater train A hardware, unavailabilities of RHR due to maintenance and, failure of the operators to recover RHRSW are significant contributors to core damage.

B.2.11.3 Application of Recovery Factors

Given the significant time available for initiation of decay heat removal, approximately two days, restoration/repair was credited for Class 2 sequences. These recovery actions were only applied to random failures or unavailabilities due to maintenance and testing and were not applied to equipment failed by the fire. In addition to the crediting of restoration/repair for Class 2 sequences, credit for restoration was applied to a restricted set of Class 1D cutsets. Manual alignment of makeup to the condenser was credited for Class 1D cutsets because of the considerable time (approximately 6 hours) available to perform this task. Restoration was also credited for events where HPCI or RCIC were initially successful but later failed because circuit breakers opened and interrupted power to the station batteries. Many hours are available to perform this operation.

The probability of recovering or repairing equipment was determined using the time available to recover the equipment prior to core damage and an assumed log mean time required to perform the repair. The time available to recover the component was derived from Modular Accident Analysis Program (MAAP) analyses performed in support of the internal events PRA. A non-recovery probability was then generated based on a simple exponential model for repair and a mean time to recover mechanical and electrical equipment from WASH-1400 [9]. Only one recovery/repair factor was applied to each cutset.

Table B.2.11.1 Monticello Plant Response to Area-Specific Fires

IPEEE Fire Sub-area	Fire Zone Description	Trains/functions failed by fire.	Train/function Available (not affected by fire; cables tracked)	Ignition Frequenc y	Class 1A	Class 1D	Class 2	CDF	Comments
I/1B	896' SE RHR and CS Room - DIV I	RCIC, CS-I, LPCI-I, SPC-I	FW, HPCI, SRV, LPCI-II, CS-II, SPC-II, HPV	3.49E-3	2.74E-9	<1E-9	3.89E-8	4.71E-8	CDF possibly reduced by crediting CRD or manual suppression.
I/2B	935' CRD Hydraulic Control Area	MCC-11, RCIC, LPCI-I, CS-I, SPC-I,	FW, HPCI, SRV, CS-II, LPCI-II, SPC-II, HPV	5.74E-3	6.75E-9	<1E-9	9.26E-8	9.93E-8	CDF possibly reduced by crediting CRD or manual suppression.
I/2E	935' Northeast T.I.P. Room	RCIC	FW, HPCI, SRV, LPCI, CS, SPC-I, HPV	1.00E-4	<1E-9	<1E-9	<1E-9	<1E-9	No spread of fire into or out of room.
I/2G	935' East Shutdown Cooling Room	LPCI-I	FW, RCIC, HPCI, SRV, CS, LPCI-II, SPC, HPV	1.00E-4	<1E-9	<1E-9	<1E-9	<1E-9	No spread of fire into or out of room.
I/3B	962' MCC & Standby Liquid Control Sys Area	MCC-11, LPCI-I, CS-I, SPC-I	FW, RCIC, HPCI, SRV, CS-II, LPCI-II, SPC-II, HPV	9.98E-3	1.51E-9	1.32E-8	1.44E-7	1.58E-7	CDF possibly reduced by crediting CRD or manual suppression.
I/BS8 (Zones 3E, 4A-E, 5A-C, 6)	985' Equipment Hatch Area	MCC-32	FW, RCIC, HPCI, SRV, CS, LPCI, SPC, HPV	3.09E-2	4.80E-9	4.58E-8	2.03E-8	7.09E-8	CDF possibly reduced by crediting CRD or manual suppression.
II/1A	896' SW RHR & CS Pump Room - Div II	LPCI-II. CS-II, SPC-II	FW, RCIC, HPCI, SRV, CS-I, LPCI-I, SPC-I, HPV	3.90E-3	<1E-9	<1E-9	5.67E-9	6.20E-9	CDF possibly reduced by crediting CRD or manual suppression.
II/2F	935' Steam Chase	HPCI, RCIC	FW, SRV, CS, LPCI, SPC, HPV	6.60E-4	5.18E-9	<1E-9	9.33E-9	5.78E-9	CDF possibly reduced by crediting CRD

Table B.2.11.1 (Continued) Monticello Plant Response to Area-Specific Fires

IPEEE Fire Sub-area	Fire Zone Description	Trains/functions failed by fire.	Train/function Available (not affected by fire; cables tracked)	Ignition Frequency	Class 1A	Class 1D	Class 2	CDF	Comments
IV/1F	896' Suppression Pool Area	CS-I, SPC-I, HPV	FW, RCIC, HPCI, SRV, CS-II, LPCI-II, SPC-II	3.80E-4	<1E-9	<1E-9	2.92E-9	2.92E-9	Fire assumed to fail only a single division of CS/LPCI Separation between divisions & no intervening combustibles.
IX/12A	911' Lower 4KV Area	BUS11/13, EDG11, LC101/103/109, 125VDC#111, SWP-11, Air Compr #11, RHRSW-I, FW-A, CRD-A, RCIC (Long Term), LPCI-I, CS-I, COND-A, SPC-I	FW-B, HPCI, CRD-B, LPCI-II, CS-II, SPC-II, SRV, HPV	5.35E-3	3.68E-7	1.29E-7	6.38E-9	5.03E-7	CDF possibly reduced by crediting manual suppression, and/or considering switchgears as "virtual" rooms.
IX/23A	919' Intake Structure Pump Room	MCC-13/23, RHRSW-II, EDGSW-II, EDG12	FW, RCIC, HPCI, SRV, CS, LPCI, SPC-I, HPV	1.02E-2	7.36E-9	<1E-9	1.26E-8	2.00E-8	CDF possibly reduced by crediting CRD or manual suppression.
V/3A	Recirc MG Room	SDC, 125VDC Pnl#313	FW, RCIC, HPCI, CS, LPCI, SRV, SPC, HPV	1.02E-2	1.51E-9	<1E-9	6.47E-9	7.98E-9	CDF possibly reduced by crediting CRD or manual suppression.

Table B.2.11.1 (Continued) Monticello Plant Response to Area-Specific Fires

IPEEE Fire Sub-area	Fire Zone Description	Trains/functions failed by fire.	Train/function Available (not affected by fire; cables tracked)	Ignition Frequency	Class 1A	Class 1D	Class 2	CDF	Comments
VI/8 **	939' Admin Cable Spreading Room	If general area fire is not suppressed, all systems not controlled from ASDS panel are assumed failed.	Operation of Div II of CS, RHR (SPC), 4/8 SRVs, HPV and #12 DG is possible from the ASDS panel.	4.40E-3 (Total) 1.20E-3 (General Area)	4.36E-7	9.89E-7	2.43E-8	1.45E-6	Fire damage is limited to FW cables if automatic suppression functions. See section B.2.10 for details.
VII/7C	928' Admin Battery Room, D-2, D-4	HPCI, CS-II, LPCI-II, SPC-II, BATTERY 12	FW, RCIC, SRV, CS-I, LPCI-I, SPC-I, HPV	1.13E-3	1.98E-9	<1E-9	4.39E-9	6.37E-9	CDF possibly reduced by crediting CRD, or manual suppression.
VIII/9 **	Admin Control Room	If fire is not suppressed, all systems not controlled from ASDS panel are assumed failed.	Operation of Div II of CS, RHR (SPC), 4/8 SRVs, HPV and #12 DG is possible from the ASDS panel.	9.76E-03	4.37E-7	9.89E-7	2.43E-8	1.45E-6	If manual suppression succeeds, fire damage is limited to initiating panel. See section B.2.10 for details.
X/BS9 Zones 12B-E, 14B, 14C, 30	Turbine Bldg Turbine Deck	FW/COND, HPCI, MCOND	RCIC, CRD, SRV, CS, LPCI, SPC, HPV	1.21E-2	8.93E-8	3.9E-8	2.39E-9	1.31E-7	CDF possibly reduced by crediting manual suppression.
XII/14A	931' Upper 4KV Area	SWP-12/13, HPV HPCI, CRD-B, RBCCW-B, COND, EDG12, RHRSW-II, LC102/104, CS-II, SPC-II,	FW/COND, RCIC, CRD-A, CS-I, LPCI-I, SPC-I, SRV	3.70E-4	3.40E-8	2.64E-8	1.70E-8	7.75E-8	Fire in TB Load Center #2 that starts outside of cabinets and engulfs the entire room. CDF possible reduced by manual suppression.
XII/14A-102	931' Upper 4KV Area	480 VAC LOAD CENTER 102	FW, RCIC, HPCI, CRD, SRV, CS, LPCI, SPC, HPV	6.10E-4	<1E-9	<1E-9	<1E-9	<1E-9	Fire in Load Center 102 cabinet.

Table B.2.11.1 (Continued) Monticello Plant Response to Area-Specific Fires

IPEEE Fire Sub-area	Fire Zone Description	Trains/functions failed by fire.	Train/function Available (not affected by fire; cables tracked)	Ignition Frequency	Class 1A	Class 1D	Class 2	CDF	Comments
XII/14A-104	931' Upper 4KV Area	480 VAC LOAD CENTER 104	FW, RCIC, CRD-A, SRV, CS-I, LPCI-I, SPC-I	6.10E-4	4.13E-8	1.31E-8	2.26E-8	7.70E-8	Fire in Load Center 104 cabinet.
XII/14A-16	931' Upper 4KV Area	4KV BUS #16 SWITCHGEAR	FW, RCIC, CS-I, CRD-A, LPCI-I, SRV	5.40E-4	3.64E-8	2.98E-8	2.29E-8	8.91E-8	Fire in switchgear #16.
XII/14A-14	931' Upper 4KV Area	4KV BUS 14 SWITCHGEAR	FW, RCIC, HPCI, CRD, SRV, CS, LPCI, SPC, HPV	5.40E-4	<1E-9	<1E-9	<1E-9	<1E-9	Fire in switchgear #14.
XII/14A-21	931' Upper 4KV Area	MCC-121	FW, RCIC, HPCI, CRD, SRV, CS, LPCI, SPC, HPV	2.40E-3	<1E-9	<1E-9	<1E-9	<1E-9	Fire in MCC-21.
XIII/15A (Includes XV/15C)	931' Standby Diesel Generator #12 Room	EDG 12	FW, RCIC, HPCI, CRD, SRV, CS, LPCI, SPC, HPV	2.92E-2	2.19E-8	<1E-9	4.85E-8	7.04E-8	CDF possibly reduced by crediting suppression.
XIV/15B (Includes XVI/15D)	931' Standby Diesel Generator #11 Room	EDG 11	FW, RCIC, HPCI, CRD, SRV, CS, LPCI, SPC, HPV	2.91E-2	2.20E-8	<1E-9	2.77E-8	4.97E-8	CDF possibly reduced by crediting suppression.
XXI/31A	EFT 1st Floor, DIV I	EDG 11 Rm Fan, MCC-33, PNL Y71, RCIC, RHRSW-I	FW, HPCI, SRV, CS, LPCI, SPC-II, HPV	2.90E-4	<1E-9	<1E-9	3.68E-9	3.97E-9	CDF possibly reduced by crediting CRD, WW/DW vents or manual suppression.
XXI/32A	EFT 2nd Floor, DIV I	MCC-34, PNL Y71, RCIC, RHRSW-I, SPC-I	FW, HPCI, SRV, CS, LPCI, SPC-II, HPV	1.99E-3	2.56E-9	<1E-9	9.69E-9	1.22E-8	CDF possibly reduced by crediting CRD, WW/DW vents or, manual suppression.

Table B.2.11.1 (Continued) Monticello Plant Response to Area-Specific Fires

IPEEE Fire Sub-area	Fire Zone Description	Trains/functions failed by fire.	Train/function Available (not affected by fire; cables tracked)	Ignition Frequency	Class 1A	Class 1D	Class 2	CDF	Comments
XXII/33	EFT 3rd Floor Cable Tunnel & Ext Duct	EDG 12, PNL Y30, PNL Y80, HPCI, RHRSW-II, CS-II, SPC-II	FW, RCIC, SRV, CS-I, LPCI-I, SPC-I, HPV	9.20E-4	8.32E-9	<1E-9	3.32E-8	4.15E-8	CDF possibly reduced by crediting CRD, WW/DW vent or, manual suppression.
II/BS1 Zones 1D, 1E, 1G	NW Corner of Rx Bldg Lower Level	HPCI	FW, RCIC, SRV, CS, LPCI, SPC, HPV	9.00E-3	5.53E-9	<1E-9	7.59E-9	1.31E-8	CDF possibly reduced by crediting CRD or manual suppression.
II/BS2 Zones 2C, 2H, 3C, 3D	West Side of Rx Bldg 935/962' Levels	MCC-22, ECCS Logic, HPCI, RCIC, CS-II, LPCI-II, SPC-II, HPV	FW, CS-I, LPCI-I, SPC-I, SRV	1.87E-2	2.03E-8	1.22E-8	5.23E-7	5.56E-7	CDF possibly reduced by crediting CRD or suppression.
III/BS3 Zones 1C, 2A	NE corner of Rx Bldg 896/935' Levels	HPCI, RCIC, CS-II, LPCI-II, SPC-II	FW, SRV, CS-I, LPCI-I, SPC-I, SRV-I, SRV-II, HPV	8.01E-3	6.29E-8	1.42E-7	1.32E-8	2.18E-7	CDF possibly reduced by crediting CRD or manual suppression.
IX/BS4 Zones 13B, 13C, 16, 19C, 23B, 37	MCC 133/ Feedwater Pump Area (908/911' Level)	AC #11, EDG11, MCC-12/23/31/ 33/34, PNL *11, SPC-I RBCCW-I, SWP 11, FW, RCIC, CRD, CS-I, LPCI-I,	HPCI, SRV, CS-II, HPV, LPCI-II, SPC-II	1.60E-2	5.09E-7	6.45E-7	4.63E-8	1.20E-6	CDF possibly reduced by 1) containment venting (WW/DW) and/or, 2) crediting manual suppression.
XII/BS5 Zones 17, 19A, 19B	MCC 142/143, TB Fire Area XII, 931' Level	RBCCW-B, HPV RHRSW-II, PNL Y30, EDG12, MCC-21/42/43/44, FW, HPCI, CRD-B, CS-II, LPCI-II, COND, SPC-II,	RCIC, CS-I, CRD-A, HPV, LPCI-I, SPC-I, SRV	6.62E-3	6.40E-7	4.55E-7	1.77E-7	1.27E-6	CDF possibly reduced by 1) containment venting (WW/DW) and/or, 2) crediting manual suppression.

Table B.2.11.1 (Continued) Monticello Plant Response to Area-Specific Fires

IPEEE Fire Sub-area	Fire Zone Description	Trains/functions failed by fire.	Train/function Available (not affected by fire; cables tracked)	Ignition Frequency	Class 1A	Class 1D	Class 2	CDF	Comments
XXII/BS6 Zones 31B, 32B	EFT Bldg Fire Area XXII	125 VDC PNL#211, MCC-44, HPCI, CS-II	FW, RCIC, CS-I, LPCI-I, SRV, SPC-I, HPV	3.46E-3	3.32E-8	9.13E-8	2.80E-7	4.05E-7	CDF possibly reduced by crediting CRD, WW/DW vents or, manual suppression.
VI/BS7 Zones 7A, 7B	Battery Rooms 7A & 7B	BATTERY#13, PNL Y70, RCIC, SRV-I, CS-I, SPC-I	FW, HPCI, SRV, CS-II, LPCI-II, SPC-II, HPV	4.02E-3	2.72E-7	5.72E-9	4.29E-8	3.21E-7	CDF possibly reduced by crediting CRD or manual suppression.
MONTICELLO CDF TOTALS	N/A	N/A	N/A	N/A	2.91E-6	3.25E-6	1.65E-6	7.81E-6	

NOTES: ** Manual or automatic suppression credited.

B.2.12 Containment Performance

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

The containment at Monticello is a Mark I design and is normally inerted with nitrogen. Because the containment contains minimal combustible material and is inerted during power operation, a significant fire within the containment is not expected to occur. The spaces surrounding the containment also contain very little combustible material. The screening criteria used in the FIVE methodology indicates that a significant fire in these compartments is not likely given their combustible loading. The same methodology also indicates that fire spread between these compartments is not credible. Therefore, because any fire in the spaces adjoining the containment will be contained within a single compartment and will be of limited duration and intensity, structural damage to the containment is not expected.

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

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

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

2) Similar to the control circuits on air-operated and solenoid-operated valves, it is unlikely that a hot short in motor-operated valve circuits could occur without actuating the circuit's protective features, such as fuses. In additional, normally open motor-operated valves typically are in series with a closed isolation valve. The integrity of these piping systems would be unaffected by the fire. Low pressure systems with motor-operated valves that connect to reactor coolant piping either include at least one check valve in series with the motor-operated isolation valves

or have two normally closed motor-operated valves. Therefore, containment bypass due to fire-induced spurious operation of motor-operated valves would require concurrent piping failure, simultaneous operation of the motor-operated valve, and/or failure of other valves.

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

B.2.13 Treatment of Fire Risk Scoping Study Issues

NRC Generic Letter 88/20, Supplement 4 lists the following fire risk scoping study issues to be addressed in IPEEE fire analyses:

(1) Seismic/fire interactions,
(2) Fire barrier assessment,
(3) Effectiveness of manual fire fighting,
(4) Effects of fire suppressants on safety equipment (total environment equipment survival), and
(5) Control systems interactions.

The specific concerns regarding each of these issues are discussed in the FIVE methodology. This methodology was used as guidance for evaluating each of the issues. Where appropriate, relevant fire risk scoping study issues have been incorporated into other phases of this study, such as the area screening and the detailed fire scenario evaluation.

Review of the fire risk scoping study issues resulted in the conclusion that these issues are not significant contributors to fire-induced core damage at Monticello.

The evaluation of each fire risk scoping study issue is discussed below.

B.2.13.1 Seismic/Fire Interactions

This issue involves three concerns: seismically induced fires, seismically induced actuation of fire protection systems, and seismically induced degradation of fire suppression systems.

Seismically induced Fires

In general, earthquakes are not known to cause fires in industrial facilities [12]. However, the potential failure of vessels containing flammable or combustible liquids or gases could cause a fire hazard in the plant following an earthquake. As a part of the seismic walkdowns, a survey of tanks and vessels that may contain flammable fluids was performed.

Only one tank, the turbine lube oil tank located in the MCC 133/feedwater pump area (see burn sequence IX/BS4), was identified to have the potential for significant consequences were it to fail as a result of a seismic event. Were the tank to slide during an earthquake, piping penetrations could fail, causing lube oil to flow from the tank to the floor of this area. However, dikes on the floor of this room would contain much or all of the oil that would drain from this tank. Further, drains to the radwaste receiver tanks would remove much of the oil from the room. The concrete walls surrounding this area in conjunction with the limited volume of oil in the tank ensures that oil will not spread out of the area.

Ignition sources such as pump and air compressor motors are elevated on pedestals above the floor, which would limit the potential for starting a fire. Even if the oil were to be ignited and affect operation of the Division I 480V cabeling in the room, alternate equipment powered from Division II switchgear remains available to provide adequate core cooling. For these reasons, additional analysis of this event is considered unwarranted.

Seismic Actuation of Fire Suppression Systems

The NRC's information notice 94-12 notes that (1) mercury relays are susceptible to seismic actuation, (2) smoke detectors could be actuated by dust rising during a seismic event, and (3) unprotected essential components could be damaged by spray from deluge systems. Mercury relays and fire suppression equipment actuated by smoke detectors are not used for fire protection of essential equipment considered in the Monticello IPEEE.

Of the plant areas containing essential equipment considered in the seismic IPEEE, only the diesel generator, the diesel generator day tank rooms, and the intake structure are protected by fire water systems. Loss of essential equipment in these areas due to inadvertent actuation of the fire water system by a seismic event is considered unlikely. These areas are protected by pre-action deluge systems. Actuation of a system requires two events: opening the deluge valves by a signal from the thermal detectors and melting the fusible links in the sprinkler heads. Occurrence of both events due to an earthquake is unlikely. In addition, the motors for RHR service water pumps P-109A through D and emergency diesel generator service water pumps P-111A and B are protected by spray shields.

Seismic Degradation of Fire Protection Systems

During an earthquake, fire suppression systems could disable nearby safe shutdown components either by colliding with them, or by bursting and either spraying or flooding the equipment. Such interactions were investigated as a part of the seismic walkdowns. Spray or flooding of essential components due to actuation of fire water systems in the diesel generator building or the intake structure is considered unlikely, as discussed above. No potential for failure of essential equipment due to collision with fire protection systems was identified in the seismic walkdown.

B.2.13.2 Fire Barrier Effectiveness

Fire barriers are used at Monticello to provide physical separation of redundant trains of safe shutdown equipment. Qualification of these barriers must be maintained to ensure an effective fire protection program. A series of detailed barrier inspection procedures are implemented to inspect all fire area boundaries and certain fire zone boundaries for the express purpose of protecting safe shutdown equipment. Fire barrier inspection procedures require that every square inch of every boundary be inspected, including penetration seals and fire dampers. Fire doors are inspected and maintained per procedure. All fire barrier inspections are performed on an eighteen-month interval.

In addition to inspection of the fire area boundaries required by Appendix R, certain fire zone boundaries are also inspected per previous NRC commitments and/or good fire protection practices. Other fire barrier concerns such as fire damper operability, as outlined in NRC information notices 83-69 and 89-52, have been resolved with walkdowns or inspections, or by modifying the operating procedures. This detailed inspection and maintenance program ensures that all fire boundaries are adequate and in good repair. Fire barrier effectiveness is ensured by implementation of these procedures.

B.2.13.3 Effectiveness of Manual Fire Fighting

NUREG/CR-5088, "Fire Risk Scoping Evaluation" [11], identified six components of an effective manual fire fighting program: (1) fire reporting, (2) fire brigade personnel and equipment, (3) fire brigade training, (4) fire brigade practice, (5) fire brigade drills, and (6) record keeping on fire brigade members. NSP's fire fighting procedures, training, and administrative work instructions address all six of these issues.

Fire reporting is accomplished with two-way radios carried by the operators and staged with the fire brigade equipment, or via two phone lines designated for that purpose. Use and staging of this equipment is detailed in plant procedures. Adequate staffing consists of six operating shift fire brigades of five people each. Three of the members must be operations personnel. Supporting equipment is prestaged in the fire brigade equipment room and includes personal protective equipment, communications equipment, portable lights and ventilation, etc.

Course work associated with fire brigade training covers subjects ranging from basic principles of fire chemistry and physics to more advanced subjects including evaluation of fire hazards and fighting fires in confined areas. All fire brigade members receive hands-on fire fighting training at least once per year to provide experience in actual fire extinguishment and the use of emergency breathing apparatus. Fire brigade drills are performed in the plant so that each fire brigade shift can practice as a team. Backshift drills and unannounced drills are performed for each brigade at least once per year.

Detailed training records and periodic quality assurance audits of the fire protection program assess the adequacy of the fire brigade training. Audit reports are kept on file at the plant.

Based on an examination of Monticello's established fire fighting training program, the attributes of an adequate fire protection program related to manual fire fighting identified in NUREG/CR-5088 are satisfied. The plant's fire brigade and manual fire fighting capability is therefore considered to be effective. Section B.2.8 describes how manual fire fighting is accounted for in this study.

B.2.13.4 Total Environment Equipment Survival

This issue includes the following three concerns:

a) The potential for adverse effects on plant equipment caused by combustion products released from the fire causing damage to, and possible loss of, safe shutdown function.

b) The spurious or inadvertent actuation of fire suppression systems resulting in the loss of safe shutdown functions.

c) Operator effectiveness in performing manual safe shutdown actions and potentially misdirected suppression effects in smoke-filled environments.

With the exception of the control and cable spreading rooms, all fire initiators included in the accident sequence quantification are assumed to spread and engulf the entire sub-area in which they are assumed to occur. Smoke effects on equipment located in these spaces is not an issue because the equipment is assumed to be destroyed by the fire. Equipment in adjoining spaces is unlikely to be damaged because the barriers that prevent the fire spread will also limit smoke propagation. Smoke that does propagate to other spaces will be dissipated. In addition, the FIVE methodology does not currently evaluate non-thermal environmental effects of smoke on equipment because the detrimental effects of smoke on equipment are not believed to be significant.

Use of automatic water fire suppression systems at the Monticello station is limited. This type of system is located over equipment in the diesel generator rooms, the feedwater pump area, and the intake structure. Effects of an inadvertent actuation are minimized by installation of spray shields protecting the RHR service water pumps located in the intake structure, fusible links installed in the sprinkler heads located over the #12 feedwater pump, and the use of pre-action water suppression systems for the diesel generators and intake structure. Susceptibility of multiple trains of safe shutdown equipment to spurious actuation of suppression systems is not expected in any case.

Manual actions to operate equipment outside of the control room or ASDS panel are not given significant credit in this study. Manual response to fires inside the control room is discussed in Section B.2.8. Review of the heating, ventilation, and air conditioning systems determined that sufficient ventilation is available to prevent excessive smoke propagation between systems and structures. Emergency lighting is positioned throughout the plant and self-contained breathing apparatus equipment is also staged at appropriate locations in the plant. This equipment allows the fire fighter to effectively combat any fire.

B.2.13.5 Control Systems Interactions

Control system interactions following a fire is principally a concern at facilities without a remote shutdown capability. Installation of the alternate shutdown system panel resolved this issue at the Monticello station. This panel allows the operators to remotely control one train of safe shutdown equipment from the EFT Building.

One of the primary features of this panel is that cables supporting Division II components can be completely isolated from the control and cable spreading rooms. This feature allows remote operation of the equipment regardless of the condition of these rooms. The Monticello operations manual provides the necessary guidance to control the plant from this panel. In addition to the written guidance, all tools and equipment required to implement the actions are staged near the panel.

B.2.14 USI-A45 and Other Safety Issues

Unresolved safety issue USI-A45, "Shutdown Decay Heat Removal Requirements," addresses the adequacy of the heat removal function at operating plants. For the purposes of the IPEEE, decay heat removal is defined as successful removal of heat from the containment. This definition corresponds to the "W" heading in the Monticello event trees and is identical to that used in resolving USI-A45 for the internal events PRA.

There are a number of methods of decay heat removal at Monticello. These consist of the main condenser, containment venting, the RHR system in either suppression pool cooling, wetwell spray, drywell spray, or shutdown cooling modes, and the reactor water cleanup system.

The main condenser is the preferred means of removing decay heat during normal shutdown until reactor pressure drops to the point where RHR shutdown cooling can be placed in service. Important support system requirements for the main condenser include offsite power, circulating water, condensate, instrument air and service water. To limit cable tracking, the main condenser was not credited in the fire IPEEE.

If the main condenser is unavailable, RHR suppression pool cooling is used as an indirect decay heat removal system, removing heat from the reactor vessel via the SRVs and suppression pool cooling. Suppression pool cooling is the principal mode of RHR containment heat removal credited in the IPEEE. Other operating modes of RHR which can remove decay heat include shutdown cooling, wetwell sprays and drywell sprays. Shutdown cooling can remove decay heat once reactor pressure has been lowered. Wetwell or drywell sprays are initiated at high containment pressures and temperatures. Due to common components required for all modes of RHR decay heat removal, the fire IPEEE accident sequence quantification credited only the suppression pool cooling and shutdown cooling modes of RHR.

If all modes of RHR were unavailable in addition to the loss of the main condenser, containment pressure would gradually increase. One to two days would be required to pressurize containment to design pressure, depending on whether makeup to the reactor was from the suppression pool or from sources external to the containment. Recovery actions would be underway to correct existing failures. Containment sprays would be used to mitigate the rise in containment pressure. If containment sprays and recovery actions were all unsuccessful, the containment would be vented prior to reaching 56 psig. To limit the amount of cable tracking, only the hard pipe vent was credited as a means of controlling containment pressure in the fire IPEEE.

If all decay heat removal systems including venting failed, decay heat would cause the containment pressure to increase slowly toward the ultimate containment pressure. Upon exceeding 70 psig, the differential pressure between containment and the pneumatic supply to the

SRVs is so low that the SRVs cannot be opened to depressurize the reactor. Low pressure systems are assumed to be unable to inject to the reactor as the reactor pressure eventually exceeds their shutoff pressures. Two to three days are necessary to pressurize the containment to its ultimate capacity of approximately 103 psig. At this point the containment is assumed to fail at the drywell head or torus expansion bellows. It is assumed that the steam release rate is small enough that containment depressurization to less than 70 psig does not occur. Low pressure systems are assumed to remain unavailable under these conditions. HPCI, feedwater and CRD are therefore the only sources of makeup credited for a loss of containment pressure control.

Actions which could significantly prolong the event without necessarily preventing fuel damage were also not credited. Those actions included use of the reactor water cleanup system, either to remove heat through the non-regenerative heat exchanger or to blow down to radwaste, or by use of an external source such as the fire system for containment sprays, which would increase the water mass in the torus.

The decay heat removal issue was examined as part of the IPE, with the details documented in sections 3.4.4 and 6.6 of the IPE [2]. This examination indicates that the failure of decay heat removal capability does not contribute significantly to the potential for core damage. An analysis of how fire hazards affect the ability to remove decay heat from containment is covered in Section B.2.11.1 under the discussion of accident Class 2. The results of this analysis do not differ from those in the IPE.

Monticello design features for decay heat removal during fires include the main condenser, four modes of RHR, and containment venting. Loss of containment heat removal sequences due to fire only contribute about 1.65E-6 per year to the core damage frequency. These redundant and diverse systems for decay heat removal are adequate to resolve this generic issue.

B.2.15 Results and Conclusions

B.2.15.1 Summary of Results

The total plant risk due to fires at Monticello is calculated to be less than 7.8E-6 core damage events per year. These results are summarized by fire area in Table B.2.11.1. Eighty three percent of the plant risk associated with internal fires can be traced to seven rooms/burn sequences. These rooms/burn areas consist of (1) the main control room, (2) turbine building 931'(fire zones XII/17, 19A and 19B), (3) the MCC 133/feedwater pump area, (4) the cable spreading room, (5) the reactor building 935/962' west, (6) the lower 4KV switchgear room and, (7) the Division II area of the EFT building.

B.2.15.2 Conclusions and Recommendations

The results of the Fire IPEEE accident sequence quantification were derived using a methodology that includes a number of conservative assumptions. Fires were assumed to increase until they completely engulfed the sub-area in which they started. In addition, with the exception of the main control room, the cable spreading room and, the feedwater pump area, the effects of fire suppression were not credited. Recovery actions were only applied to accident sequences that allow many hours for recovery, and then only to those components not located in the area affected by the fire. When the recovery actions were applied, the recovery of only a single failed component was allowed even if there were multiple failures to which recovery factors could be applied. Therefore this methodology, while demonstrating relatively low risk due to internal fires, yields conservative core damage frequencies.

The relatively low plant risk due to fires is in large part due to Monticello's implementation of the requirements of 10CFR50, Appendix R. These requirements, including separation of alternate or redundant trains of safe shutdown equipment, installation of fire barriers, and an installation of an alternate shutdown system outside the control and cable spreading rooms, combine to limit the total risk due to fires. The administrative control of transient combustibles also contributes to the low fire risk.

Core damage will not occur unless random equipment failures unrelated to damage caused by the fire also occur. This fact, in conjunction with the low overall core damage frequency due to fires, precluded identification of any risk significant insights. However, one potential improvement to this study was noted. This improvement addresses conservatisms in the current analysis. Because of the location specific nature of the analyses and the effort involved in performing these analyses for all areas throughout the plant, manual suppression effects were analyzed and credited only in the control room. If future applications warrant, the effects of manual suppression can be analyzed and credited on a location by location basis. Similarly, CRD and the main condenser were only credited for fires in a few areas. These systems may be credited on a location by location basis by verifying that cables/equipment required by these systems are not located in the locations in question. Incorporation of this improvement may result in a significant reduction in overall calculated plant risk. These calculational conservatisms should be considered before considering any plant modification.

B.2.16 References

1. Generic Letter No. 88-20, Supplement 4, "Individual Plant Examination Of External Events (IPEEE) for Severe Accident Vulnerabilities", United States Nuclear Regulatory Commission, June 1991.

2. Monticello Nuclear Generating Plant Individual Plant Examination (IPE), NSPNAD-92003, Rev. 0, Northern States Power Company, February 1992.

3. Safe Shutdown Analysis Engineering Report - Monticello Nuclear Generating Plant, Rev.2, Northern States Power Company, February 1991.

4. Fire Induced Vulnerabilities Evaluation (FIVE) Plant Screening Guide, EPRI, September 1991.

5. NUREG/CR-4527/1 of 2, "An Experimental Investigation of Internally Ignited Fires In Nuclear Power Plant Control Cabinets: Part 1: Cabinet Effect Tests," U.S. Nuclear Regulatory Commission, April 1987.

6. Letter dated December 21, 1989, from Frank McGinnis of Engineering Planning and Management, Inc., to Mr. Greg Olson of Northern States Power, "Generic Letter 86-10 Evaluations."

7. Updated Fire Hazards Analysis - Monticello Nuclear Generating Plant, Rev.7, Northern States Power Company, January 1990.

8. "Fire Events Database for U.S. Nuclear Power Plants," NSAC-178L, June 1992.

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

10. "Report on Full-Scale Horizontal Cable Tray Fire Tests," FNAL-TM-1549, Fermi National Accelerator Laboratory, September 1989.

11. NUREG/CR-5088, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," NRC, January 1989.

12. EPRI NP-6041, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Rev. 1, EPRI, July 1991.