ERI/NRC 95-513

TECHNICAL EVALUATION REPORT ON THE "SUBMITTAL-ONLY" REVIEW OF THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS AT NINE MILE POINT NUCLEAR STATION, UNIT 2

FINAL REPORT

Completed: December 1996
Final: March 1998

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Work Performed Under the Auspices of the
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. 20555
Contract No. 04-94-050

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Enclosure 2

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Principal Investigator

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EXECUTIVE SUMMARY

This technical evaluation report (TER) documents a "submittal-only" review of the individual plant examination of external events (IPEEE) conducted for the Nine Mile Point Nuclear Station, Unit 2 (NMP-2). This technical evaluation review was performed by Energy Research, Inc. (ERI) on behalf of the U.S. Nuclear Regulatory Commission (NRC). The submittal-only review process consists of the following tasks:

- Examine and evaluate the licensee's IPEEE submittal and directly relevant available documentation.
- Develop requests for additional information (RAIs) to supplement or clarify the licensee's IPEEE submittal, as necessary.
- Examine and evaluate the licensee's responses to RAIs.
- Conduct a final assessment of the strengths and weaknesses of the IPEEE submittal, and develop review conclusions.

This TER documents ERI's qualitative assessment of the NMP-2 IPEEE submittal, particularly with respect to the objectives described in Generic Letter (GL) 88-20, Supplement No. 4, and the guidance presented in NUREG-1407.

The NMP-2 IPEEE was performed by Niagara Mohawk Power Corporation (NMPC), the plant licensee, with assistance of contractor personnel. This IPEEE is a new analysis, and considers seismic; fire; and high winds, floods, and other (HFO) external initiating events. Seismic margin assessment (SMA) methodology was applied to the analysis of seismic initiating events. A combination of the Fire Induced Vulnerability Evaluation (FIVE) methodology and probabilistic risk assessment (PRA) methods was applied to the analysis of fire initiating events. HFO events were evaluated using the progressive screening approach identified in NUREG-1407; all HFO events were screened out based on conformance with the NRC Standard Review Plan (SRP).

Licensee's IPEEE Process

NMPC performed a new analysis to satisfy the GL 88-20 objectives for the NMP-2 IPEEE. No analytical freeze date was identified in the IPEEE submittal.

For the seismic IPEEE, the licensee determined high-confidence of low probability of failure (HCLPF) capacity values for components in two safe shutdown paths. The effort of determining HCLPF capacities was extended to the evaluation of seismic fragilities, since most of the work necessary to evaluate fragilities was already completed in the HCLPF determinations. A review level earthquake (RLE) of 0.5g was conservatively used for screening, rather than the value of 0.3g recommended in NUREG-1407. In addition, a seismic PRA was performed. For quantification of the seismic PRA results, the licensee employed the seismic hazard results prepared by the Electric Power Research Institute (EPRI) and by

Lawrence Livermore National Laboratory (LLNL). The seismic PRA employed a surrogate element which represented the structures, systems, and components that were screened out at a value of 0.5g peak ground acceleration (PGA).

For the fire IPEEE, the licensee used a combination of FIVE and PRA methodologies. The fire analysis consisted of the following four steps:

- qualitative screening
- quantitative screening
- fire damage evaluation screening
- fire scenario evaluation and quantification

For the evaluation of HFO events, the licensee adopted a progressive screening methodology (in accordance with NUREG-1407) that included the following major steps:

- List plant-specific external events
- Conduct progressive screening using compliance with the NRC SRP as review criteria
- Prepare documentation

Key IPEEE Findings

For seismic events, the NMP-2 IPEEE reported the following results:

- All structures, systems, and components in the simplified success path screened out for a HCLPF value equal to or greater than 0.5g.

- The mean seismic core damage frequency was calculated to be 2.5×10^{-7} per reactor-year (ry), using the EPRI seismic hazard results, and 1.2×10^{-6} per reactor year using the LLNL seismic hazard results.

Seismic core damage frequency (CDF) is dominated by loss of injection with early core damage, and loss of heat removal or injection with late core damage. The most significant seismic failures of components were determined to be:

- surrogate element (COMP1), consisting of all components that were screened out in the SMA
- offsite power
- high pressure nitrogen system (failure due to seismic interaction)

The most significant non-seismic basic events were found to be:

- failure of reactor core isolation cooling (RCIC)
- failure of high pressure core spray (HPCS)
- failure of shutdown cooling
- failure of Division I or Division II emergency diesel generator

Early containment failure or bypass events are dominated by the surrogate element, which includes an inherent assumption of failure to isolate the containment. This result is an artifice of the assumed HCLPF value for all screened-out components; the actual HCLPF of containment isolation-related structures, systems, and components is recognized to be much higher than the assumed value. Containment failure is dominated by station blackout scenarios with unisolated penetrations.

The total fire-induced CDF of 1.4×10^{-6} per reactor year represents the sum of the frequencies of core damage sequences from four fire event trees. The control room is the only fire area which survived screening. Only a limited number of initiating events were determined by the licensee to be applicable/relevant to the fire CDF quantification. These are:

- General Plant Transient
- Loss of Offsite Power
- Station Blackout
- Stuck-Open Safety Relief Valve (SRV)
- Total Loss of Service Water

The following containment failure modes were identified and evaluated by the licensee:

- Containment isolation/bypass
- Containment overpressure failure

The licensee concluded that fires were insignificant contributors to the above containment failure modes.

All HFO events were screened out based on conformance to the NRC SRP. However, the licensee performed a bounding analysis of tornado events. This bounding analysis was deemed to be incomplete. However, a conservative bounding assessment was performed as part of this review which found that the tornado-induced CDF is at most 2.6×10^{-7} per reactor year. Therefore, it is concluded that the licensee correctly screened tornadoes and high winds as an insignificant contributor to external events risk.

As previously noted, the licensee screened external flooding based SRP conformance. However, the licensee also included a more extended discussion of external flooding, attempting to make bounding arguments to the effect that external flooding is an insignificant contributor. While noting the screening of this event based on SRP conformance (as provided for in NUREG-1407), the licensee's bounding arguments appear to be flawed.

Generic Issues and Unresolved Safety Issues

The seismic IPEEE addressed the following generic and unresolved safety issues (GIs/USIs): USI A-17, "Systems Interactions in Nuclear Power Plants"; USI A-45, "Shutdown Decay Heat Removal Requirements"; and the eastern U.S. seismicity issue.

For USI A-17, NMPC analyzed spatial interactions as well as interactions due to relay chatter. Seismically caused control systems interactions that can propagate via the electrical and control systems were considered, including evaluation of control system devices, relays, sensors, thermal overloads, electrical contractors, and breakers. Systems interactions were also considered during the seismic walkdown. For

USI A-45, no weaknesses were identified during the SMA analysis, with regard to decay heat removal or any other seismic issues. A plant HCLPF of at least 0.5g was determined for the residual heat removal (RHR) system and its support systems. (Note that this 0.5g plant HCLPF is only demonstrated for a period of 24 hours, in contrast with the seismic margin methodology guidelines which specify 72 hours. The 72-hour plant HCLPF is 0.23g for NMP-2.) NMPC also relied on the quantitative results of the seismic PRA to show that the seismic risk is low at NMP-2. The eastern U.S. seismicity issue was stated by NMPC to have been resolved by the IPEEE in accordance with GL 88-20, Supplement No. 4, and by the use of a 0.5g RLE for the SMA.

For fire-related events, NMPC has explicitly addressed the Sandia Fire Risk Scoping Study issues, as well as USI A-45. Control systems interactions were addressed in the context of control room fires, with the licensee concluding that all controls required for safe shutdown have transfer or isolation switches located outside the control room. Seismic/fire interactions were addressed by focusing on the potential for seismic events to cause a release of flammable or combustible liquids or gases, by evaluating the potential for seismic actuation of fire suppression systems, and by examining the potential for seismic-induced failure of fire suppression systems. Manual fire fighting effectiveness (including smoke control) was addressed by comparing the fire brigade and fire protection attributes of NMP-2 against the EPRI evaluation of the Fire Risk Scoping Study. The licensee concluded that NMP-2 meets all the attributes identified by EPRI. The potential adverse effects on plant equipment by-products (including smoke) were addressed only in terms of short term response to the fire; the operators were assumed to be able to shut down the plant without experiencing additional equipment losses due to smoke. Total environment equipment survival (including spurious operation of suppression systems) was addressed by considering the potential short-term adverse effects of combustion products on plant equipment (for the long-term, the IPEEE assumed that the operators would be able to shut down the plant without experiencing additional equipment losses due to smoke damage). Short-term damage due to smoke was assessed as being mitigated by plant-specific design features. Spurious or inadvertent fire suppression system actuation was also assessed as being mitigated by design features and by a specific architect-engineer review of historical events identified in NRC Information Notice 83-41. The issue regarding the adequacy of fire barriers was addressed only qualitatively; no fire barrier failure rates were used in the analysis. The issue regarding the adequacy of analytical tools for fire assessments was addressed by using the FIVE methodology, which had been previously approved by the NRC for fire IPEEE assessments.

In regards to USI A-45, NMP-2 relies on the HPCS, RCIC, and RHR systems for decay heat removal in response to fire events.

For HFO events, no USIs/GIs or other non-IPEEE issues have been addressed in the IPEEE submittal. The submittal states that a re-evaluation of maximum precipitation (pertaining to GI-103) was reported in the NMP-2 Updated Safety Analysis Report (USAR). No details of this re-evaluation are presented in the submittal.

Some information is also provided in the NMP-2 IPEEE submittal which pertains to generic safety issue (GSI)-147, GSI-148, and GSI-172.

Vulnerabilities and Plant Improvements

The IPEEE submittal notes that a number of plant improvements or resolution procedures were implemented in response to the IPEEE, as summarized below:

- Racks (near 2ICS*MOV129). During the walkdown, a material storage rack near the cited motor-operated valve (MOV) was identified as having the potential to fall and impact the MOV. The rack was secured.

- Electrical Panel Hoist Assemblies. It was noted that several safety-related electrical cabinets included a hoist assembly located on the top of the panel which could move and jar equipment during an earthquake. Rail stops were installed to preclude this problem from occurring.

- Control Building (seismically induced flood concern). The presence of fire-water piping in the control building was investigated during the walkdown. Detailed evaluation subsequent to the walkdown concluded that a HCLPF of 0.5g could be justified, and no modifications were found to be necessary.

- Tornado Interaction. Construction of some new non-safety-related buildings on site did not include tornado design criteria as suggested by the USAR. This situation was evaluated and found not to be risk significant. However, a plant deviation report was written to document the finding and to resolve the discrepancy between the USAR and the as-built plant.

The IPEEE submittal concludes that no vulnerabilities were found. Potential improvements in procedures and training in response to control room fires are being assessed. No new risk-significant systems were identified related to implementation of the maintenance rule. Systems identified as important in the IPEEE were already included.

Observations

For seismic events, the simplified success path used in the NMP-2 SMA does not include low-pressure makeup. This unusual result arises from a low HCLPF capacity (0.23g PGA) of the non-safety-related nitrogen bottles, which provide the long-term (72-hour) capability to operate the SRVs and to permit continuation of low-pressure makeup. (The low HCLPF capacity of the nitrogen bottle arises due to seismic interaction with other nearby tanks.) This situation was not identified by the licensee as a vulnerability because it is assumed that the plant Emergency Operating Procedures (EOPs) can be used to mitigate any accident where all structures, systems, and components not on the 0.5g success path may fail. The EOPs provide direction to the operators to use HPCS and RCIC for inventory control, and to use the RHR system in the suppression-pool cooling mode. The licensee is not aware of any EOP or stipulation in the Boiling Water Reactor (BWR) Owners Group Emergency Procedure Guideline (EPG) that must be violated in order to maintain the plant for 72 hours using only the structures, systems, and components on the 0.5g simplified success path. While not credited in the SMA or the seismic PRA, the nitrogen system can be used to support low-pressure injection for at least 24 hours following an earthquake. The licensee states that, with mitigation for 24 hours, decay heat loads would be sufficiently low to permit time for recovery actions. Finally, there are two success paths which have a HCLPF of 0.23g which can maintain the plant in safe shutdown for 72 hours. These success paths are not credited in the SMA. The

licensee states that the PRA shows the nitrogen system failure to be of limited risk significance. Although the licensee performed the SMA at a 0.5g review level and states that the plant demonstrates a 0.5g HCLPF, this is only valid for 24 hours. The licensee could only demonstrate a 72-hour HCLPF (which is the basis for SMA studies identified in EPRI-NP-6041, Rev. 1) of 0.23g.

EPRI NP-6041, Rev. 1, which contains the guidance for SMA evaluations, identifies applicable failure rates for HPCS and RCIC, individually and jointly. Even for a 24-hour time period (as opposed to the 72-hour evaluation time specified for SMA purposes), the NMP-2 failure-rate values exceed the applicable SMA guidance. For 24 hours, NMP-2 failure rates for HPCS, RCIC, and HPCS/RCIC jointly are 0.14, 0.16, and 0.0224, respectively. The licensee provided corresponding values for 72 hours of 0.29, 0.31, and 0.0899. However, the 72-hour values do not include recovery. The licensee states that recovery after 24 hours has a relatively high success probability (presumably, due to the length of time available for recovery as a result of lower decay heat levels); however, no quantitative insight into this matter was provided by the licensee.

Accordingly, it does not appear that the ERPI NP-6041, Rev. 1, guidance was followed in this instance. The safety significance of this deviation from the SMA guidance, however, is apparently very low as indicated by the seismic PRA findings and the ability of the plant to maintain shutdown for 72 hours using only structures, systems, and components that are in the 0.5g HCLPF simplified success path. Thus, although the licensee deviated from the EPRI guidelines, the licensee combined a risk evaluation and a timing evaluation to take credit for recovery.

For fire events, the licensee has conducted an extensive and detailed analysis of fire events at NMP-2. However, the review notes disagreement with the following of the licensee's analytical assumptions:

- The operator recovery probabilities for the control room fire scenarios are highly optimistic.
- The heat release rate for an electrical cabinet fire is assumed to be 65 Btu/sec, and is not representative of cabinet fire-test data.
- The potential adverse effects on plant safety-related equipment due to combustion products have not been adequately addressed.

For HFO events, the licensee screened out all such initiators based on SRP conformance. The licensee provided information about tornadoes, external flooding, and transportation and nearby facility accidents. The discussion on flooding makes it apparent that, despite SRP conformance, there are no bounding PRA arguments that can be made to dismiss external flooding as a possible contributor (due, primarily, to the existence of large uncertainties in the frequency of flooding above a critical elevation, which would cause failure of the emergency switchgear). The licensee's rationale for dismissing external flooding as a contributor is incomplete.

PREFACE

The Energy Research, Inc., team members responsible for the present IPEEE review documented herein, include:

<u>Seismic</u>

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<u>Fire</u>

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M. Khatib-Rahbar, Principal Investigator and Report Review
A. Kuritzky, IPEEE Review Coordination and Integration
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This work was performed under the auspices of the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The continued technical guidance and support of various NRC staff is acknowledged.

ABBREVIATIONS

AC	Alternating Current
ADS	Automatic Depressurization System
APRM	Average Power Range Monitor
ARI	Alternate Rod Injection
ATWS	Anticipated Transients Without Scram
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CST	Condensate Storage Tank
DC	Direct Current
EOP	Emergency Operating Procedure
EPG	Emergency Procedures Guideline (BWR Owners Group)
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
FIVE	Fire Induced Vulnerability Evaluation
FRS	Floor Response Spectrum
GI	Generic Issue
GL	Generic Letter
GSI	Generic Safety Issue
HCLPF	High Confidence of Low Probability of Failure
HEP	Human Error Probability
HFO	High Winds, Floods, and Other External Events
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
MOV	Motor-Operated Valve
NFPA	National Fire Protection Association
NMP-2	Nine Mile Point, Unit 2
NMPC	Niagara Mohawk Power Corporation
NRC	United States Nuclear Regulatory Commission
PGA	Peak Ground Acceleration
PMF	Probable Maximum Flood
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
RAI	Request for Additional Information
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel

RRCS	Redundant Reactivity Control System
SER	Safety Evaluation Report
SEWS	Seismic Evaluation Worksheets
SLC	Standby Liquid Control
SMA	Seismic Margin Analysis
SME	Seismic Margin Earthquake
SMM	Seismic Margin Methodology (alternate abbreviation to SMA)
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SRT	Seismic Review Team
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
TER	Technical Evaluation Report
USAR	Updated Safety Analysis Report
USI	Unresolved Safety Issue

1 INTRODUCTION

This technical evaluation report (TER) documents the results of the "submittal-only" review of the individual plant examination of external events (IPEEE) for the Nine Mile Point Nuclear Station, Unit 2 (NMP-2) [1]. This technical evaluation review, conducted by Energy Research, Inc. (ERI), has considered various external initiators, including seismic events; fires; and high winds, floods, and other (HFO) external events.

The U.S. Nuclear Regulatory Commission (NRC) objective for this review is to determine the extent to which the IPEEE process used by the licensee, Niagara Mohawk Power Corporation (NMPC), meets the intent of Generic Letter (GL) 88-20, Supplement No. 4 [2]. Insights gained from the ERI review of the IPEEE submittal are intended to provide a reliable perspective that assists in making such a determination. This review involves a qualitative evaluation of the licensee's IPEEE submittal, development of requests for additional information (RAIs), evaluation of the licensee responses to these RAIs, and finalization of the TER.

The emphasis of this review is on describing the strengths and weaknesses of the IPEEE submittal, particularly in reference to the guidelines established in NUREG-1407 [3]. Numerical results are verified for reasonableness, not for accuracy; however, when encountered, numerical inconsistencies are reported. This TER complies with the requirements of NRC's contractor task order for an IPEEE submittal-only review.

The remainder of this section of the TER describes the plant configuration and presents an overview of the licensee's IPEEE process and insights, as well as the review process employed for evaluation of the seismic, fire, and HFO events sections of the NMP-2 IPEEE. Sections 2.1 to 2.3 of this report present ERI's findings related to the seismic, fire, and HFO events reviews, respectively. Sections 3.1 to 3.3 summarize ERI's evaluation and conclusions from the seismic, fire, and HFO events reviews, respectively. Section 4 summarizes the IPEEE insights, improvements, and licensee commitments. Section 5 includes completed IPEEE data summary and entry sheets. Finally, Section 6 provides a list of references.

1.1 Plant Characterization

NMP-2 is a boiling water reactor (BWR), based on Generic Electric's BWR/5 design, with Mark II containment. The plant is located on the southeast shore of Lake Ontario, approximately 6.2 miles (10 kilometers) northeast of the city of Oswego, New York. The site is common with Nine Mile Point, Unit 1, which is an earlier generation BWR also operated by NMPC. The Fitzpatrick Nuclear Plant, also an earlier model BWR and operated by another utility, is located adjacent to the NMP-2 site.

The original licensed power level for NMP-2 was 3,323 MWt, but at the time of the IPEEE submittal the licensee was pursuing an increased rating of 3,467 MWt. The effect of the uprating has not been considered in the IPEEE submittal. The NMP-2 site is considered to be a rock site, and has a safe shutdown earthquake (SSE) peak ground acceleration (PGA) value of 0.15g.

1.2 Overview of the Licensee's IPEEE Process and Important Insights

1.2.1 Seismic

The licensee employed the Electric Power Research Institute's (EPRI) seismic margin assessment (SMA) methodology to analyze seismic initiating events. Although NUREG-1407 designated the plant as a focused-scope plant with a review level earthquake (RLE) of 0.3g PGA, NMPC elected to use the more conservative RLE of 0.5g PGA. A simplified success path was identified for analytical purposes, involving use of the high-pressure core spray (HPCS) and reactor core isolation cooling (RCIC) systems for inventory control, and use of the residual heat removal (RHR) system in the suppression pool cooling mode for heat removal.

The licensee extended the seismic analysis in the SMA to perform a seismic probabilistic risk assessment (PRA), using a surrogate-element modeling approach in which all structures, systems, and components in the simplified success path are modeled as a surrogate component having a high confidence of low probability of failure (HCLPF) capacity of 0.5g PGA. Seismic hazard curves prepared by EPRI and by Lawrence Livermore National Laboratory (LLNL) were used to quantify the seismic risk model.

All structures, systems, and components in the simplified success path were found to meet the 0.5g RLE. In addition, the licensee calculated low values of seismic core damage frequency, estimated at 2.5×10^{-7} and 1.2×10^{-6} per reactor-year (ry), respectively, for the EPRI and LLNL seismic hazard results. If operator action to close certain motor-operated valves outside containment is credited, less than 2% of the core damage frequency is associated with early containment failure or bypass.

1.2.2 Fire

The licensee has conducted an extensive and detailed analysis of fire events at NMP-2. The licensee has used plant data from the Appendix R effort to conduct the analysis. Overall, the licensee has concluded that there are no significant fire vulnerabilities at NMP-2. The present review has identified a few items in the fire hazard analysis that the licensee may need to re-visit to ensure that this conclusion has a sound basis. This review has concerns with the following aspects of the fire IPEEE analysis:

1. The operator recovery probabilities for the control room fire scenarios are highly optimistic.

2. The heat release rate for electrical cabinet fires is assumed to be 65 Btu/sec and is not representative of cabinet fire test data.

3. The potential adverse effects on plant safety-related equipment due to combustion products have not been adequately addressed.

4. The potential for cross-zone fire and smoke spread was not considered.

5. The only seismically induced fire sources addressed were releases of flammable or combustible liquids or gases. Weakly anchored electrical cabinets have been found to be an important seismically induced fire risk contributor [4], and therefore, should also have been considered.

1.2.3 HFO Events

The licensee screened out all HFO events on the basis of conformance of the NMP-2 design to the Standard Review Plan (SRP) criteria. This approach is consistent with the guidance in NUREG-1407.

1.3 Overview of Review Process and Activities

In its qualitative review of the NMP-2 IPEEE, ERI focused on the study's completeness in reference to NUREG-1407 guidance; its ability to achieve the intent and objectives of GL 88-20, Supplement No. 4; its strengths and weaknesses with respect to the state-of-the-art; and the robustness of its conclusions. This review did not emphasize confirmation of numerical accuracy of submittal results; however, any numerical errors that were obvious to the reviewers are noted in the review findings. The review process included the following major activities:

- Completely examine the IPEEE and related documents
- Develop a preliminary TER and RAIs
- Examine responses to the RAIs
- Finalize this TER and its findings

Because these activities were performed in the context of a submittal-only review, ERI did not perform a site visit or an audit of either plant configuration or detailed supporting IPEEE analyses and data. Consequently, it is important to note that the ERI review team did not verify whether or not the data presented in the IPEEE matches the actual conditions at the plant, and whether or not the programs or procedures described by the licensee are indeed implemented at NMP-2.

1.3.1 Seismic

In conducting the seismic review, ERI generally followed the emphasis and guidelines described in the report, *Individual Plant Examination of External Events: Review Guidance* [5], for review of a seismic margin assessment and a seismic PRA, and the guidance provided in the NRC report, *IPEEE Step 1 Review Guidance Document* [6]. In addition, on the basis of the NMP-2 IPEEE submittal, ERI completed data entry tables developed in the LLNL document entitled *"IPEEE Database Data Entry Sheet Package"* [7].

In its NMP-2 seismic review, ERI examined the following documents:

- Sections 1, 2, 3, 6, 7, and 8 of the IPEEE [1]; and
- The licensee's responses to the RAIs [8] generated as part of the initial submittal review

The checklist of items identified in Reference [5] was generally consulted in conducting the seismic review. Some of the primary considerations in the seismic review have included (among others) the following items:

- Were appropriate walkdown procedures implemented, and was the walkdown effort sufficient to accomplish the objectives of the seismic IPEEE?

- Was the development of success paths performed in a manner consistent to prescribed practices? Were random and human failures properly considered in such development?

- Were component demands assessed in an appropriate manner, using valid seismic motion input and structural response modeling, as applicable? Was screening appropriately conducted?

- Were capacity calculations performed for a meaningful set of components, and are the capacity results reasonable?

- Has the surrogate element been used in such a manner so as to not obscure dominant risk contributors and to produce a valid numerical estimate of core damage frequency?

- Was the approach to seismic risk quantification appropriate, and are the results meaningful?

- Does the submittal's discussion of qualitative assessments (e.g., containment performance analysis, seismic-fire evaluation) reflect reasonable engineering judgment, and have all relevant concerns been addressed?

- Has the seismic IPEEE produced meaningful findings, has the licensee proposed valid plant improvements, and have all seismic risk outliers been addressed?

1.3.2 Fire

During this technical evaluation, ERI reviewed the fire-events portion of the IPEEE for completeness and consistency with past experience. This review was based on Sections 1, 2, 4, 5, 6, 7, and 8 of Reference [1], and on the licensee responses to fire-related RAIs [8]. The guidance provided in References [5, 6] was used to formulate the review process and the organization of this document. The data entry sheets used in Section 5 are taken from Reference [7].

The process implemented for ERI's review of the fire IPEEE included an examination of the licensee's methodology, data, and results. ERI reviewed the methodology for consistency with currently accepted and state-of-the-art methods, paying special attention to the screening methodology to ensure that no fire scenarios were prematurely eliminated. The data element of a fire IPEEE includes, among others, such items as:

- Cable routing
- Fire zone/area partitioning
- Fire occurrence frequencies
- Event sequences
- Fire detection and suppression capabilities

The conditions described, and information provided, by the licensee were evaluated to determine their reasonableness, and their similarity with other fire studies. For a few fire zones/areas that were deemed important, ERI also verified the logical development of the screening justifications/arguments (especially in the case of fire-zone screening) and the computations for fire occurrence and CDF.

1.3.3 HFO Events

The review process for HFO events closely followed the guidance provided in the report entitled *IPEEE Step 1 Review Guidance Document* [6]. This process involved examinations of the methodology, the data used, and the results and conclusions derived in the submittal. The IPEEE methodology was reviewed for consistency with currently accepted practices and NRC recommended procedures. Special attention was focused on evaluating the adequacy of data used to estimate the frequency of HFO events, and on confirming that any analysis of SRP conformance was appropriately executed. In addition, the validity of the licensee's conclusions, in consideration of the results reported in the IPEEE submittal, was assessed. Also, in some instances, computations of frequencies of occurrence of hazards, fragility values, and failure probabilities were spot checked. Review team experience was relied upon to evaluate the reasonableness of the licensee's evaluation.

2 CONTRACTOR REVIEW FINDINGS

2.1 Seismic

A summary of the licensee's seismic IPEEE process has been described in Section 1.2. Here, the licensee's seismic evaluation is described in detail, and discussion is provided regarding significant observations encountered in the present review.

2.1.1 Overview and Relevance of the Seismic IPEEE Process

As documented in NUREG-1407, NMP-2 is binned as a focused-scope plant with a RLE of 0.3g PGA. However, NMPC performed a seismic margin assessment (SMA), using an RLE of 0.5g PGA (which is conservative with respect to the review guidance), as well as a seismic PRA. A qualitative seismic containment performance analysis was also performed, including a walkdown. Although the seismic PRA was considered in the present reviewed, the principal focus of this review was on the seismic margin analysis.

The NMP-2 seismic IPEEE process is consistent with the recommended guidelines of NUREG-1407, and the performance of a seismic PRA went beyond that guidance. The licensee notes in its IPEEE submittal that most of the work for a seismic PRA had been completed by performing the SMA. The overall seismic IPEEE methodology is appropriate and relevant to severe accident analysis and vulnerability assessment. It also appears that the licensee has had a very significant participation in the study.

2.1.2 Success Paths and Component List

"Success" is defined in the seismic IPEEE as maintaining hot shutdown conditions for at least 72 hours. (This duration exceeds the mission time of 24 hours used in the individual plant examination [IPE] and seismic IPEEE PRA.) The success path for NMP-2 is the same for both small loss-of-coolant accidents (LOCAs) and transients, consistent with the licensee's statement that success criteria are essentially the same for small LOCAs and transients. The success path was developed assuming minimal credit for operator action, in order to ensure that the identification of components proceeded in a conservative manner. (However, alignment of the RHR system for suppression pool cooling is an important required manual operation.) The success path evaluation also considered the potential effects of random (non-seismic) failures and adverse human actions. A "simplified" success path was used in the NMP-2 SMA. The success path functions and required front-line systems are identified below:

Function	Front-Line Systems
Reactivity Control	Reactor Protection System
Pressure Control	Safety Relief Valves
Inventory Control	RCIC High Pressure Core Spray (HPCS)
Heat Removal	Residual Heat Removal (RHR) Suppression Pool Cooling, A & B Trains

Support systems identified in the IPEEE submittal include the following:

- Pump room cooling for RCIC, HPCS, and diesel generators
- Service water
- Division I and II alternating current (AC) power
- Division I and II direct current (DC) power
- Automatic suction transfer from the condensate storage tank (CST) for RCIC and HPCS

Neither low-pressure coolant makeup nor containment venting are modeled in the simplified success path, due to seismic interaction problems with the nitrogen gas system (which provides the capability to maintain the automatic depressurization system [ADS] valves and certain containment isolation valves open in the long term). It should be noted that the nitrogen system was not an issue in the IPE analysis as a result of the 24-hour mission time used in the IPE (see Section 3.1.2.1.2 of the IPEEE submittal), in comparison with the 72-hour mission time used in the SMA. Although the licensee performed the SMA at a 0.5g review level and states that the plant demonstrates a 0.5g HCLPF, this is only valid for 24 hours. The licensee could only demonstrate a 72-hour HCLPF (which is the basis for SMA studies identified in EPRI-NP-6041, Rev. 1) of 0.23g.

In accordance with the EPRI NP-6041 methodology, success paths must be defined for both transient events and small LOCA events. No extended discussion is provided in EPRI NP-6041 regarding the basis for excluding medium and large LOCA events. The NMP-2 IPEEE SMA assumes that medium and large LOCAs following an earthquake have a small probability as a result of the high capacity of piping and reactor coolant pressure boundary components (see Section 3.1.2.1.1 of the IPEEE submittal). The NMP-2 success path does not explicitly address the possibility of a stuck-open safety relief valve (SRV). The NMP-2 IPE equates a stuck-open SRV with a medium LOCA, and notes that RCIC is unable to provide vessel makeup for such an event [9]. Thus, for a single stuck-open SRV, makeup capability is limited to HPCS (which has a relatively high failure rate, for 24 hours, of 0.14). For a stuck-open SRV scenario, there is no alternate success path even under NMPC's approach. That is, if a stuck-open SRV occurs, and HPCS fails, RCIC is unable to provide adequate makeup and there is no alternate success path. The same consequence is true for large LOCAs for which HPCS operation is a successful outcome, but for which RCIC is unable to provide adequate makeup.

The NMP-2 IPE states that, in a general transient which does not involve vessel isolation, as many as five SRVs could open due to a pressure transient [9]. In the case of the RLE, vessel isolation would occur, and even more SRVs may likely open. Clearly, the potential exists for the occurrence of one or more stuck-open SRVs during an RLE event. For the case of five SRVs that may open during a general transient, the conditional probability of one SRV remaining open is multiplied by five to obtain an approximation to the probability of one or more stuck-open SRVs. The conditional probability for one SRV sticking open on demand is 3.16×10^{-3} per demand (see Table 3.3.1-1, Reference [9]); multiplied by five for five challenges, this yields a conditional probability of 1.6% of at least one stuck-open SRV, given a transient with the vessel not isolated. For a vessel isolation transient (which would occur in the RLE), even more SRVs would be challenged, and the conditional probability of one or more SRVs being stuck-open would rise accordingly. Thus, it is considered important for the NMP-2 success path to explicitly address medium and large LOCAs.

Additional specific comments concerning the treatment of reactivity control and reactor pressure vessel (RPV) inventory control, in success-path development, are provided below.

Reactivity Control

Reactivity control in the simplified success path for NMP-2 is always via the reactor protection system (RPS)/scram function. Under the NMP-2 SMA, there is no "alternate" success path to the RPS/scram function. A NUREG-1150 analysis indicates that approximately two-thirds of all RPS/scram failures are electrical in nature, and can be recovered by operator action (initiating a confirmatory manual scram by placing the mode switch in shutdown, which actuates the scram by alternate means). (Additionally, there are four individual push-buttons, one for each RPS channel, which can be pushed to de-energize the channels [9].)

The design of NMP-2 includes, in compliance with the NRC's rule on anticipated transients without scram (ATWSs), a standby liquid control (SLC) system which can provide reactivity control under circumstances where the normal shutdown system fails to function. No explanation is provided in the licensee's IPEEE submittal concerning the seismic capacity of the SLC system. This omission is particularly important in view of probabilistic risk assessment, containment loads, and severe-accident calculations for BWR units with Mark II containments, where there has been identified a direct link between failure to achieve shutdown and the potential for early containment failure.

The guidance in EPRI SMA methodology indicates that the SLC system should not be considered as an alternate success path due to concerns over stressful operator actions required to initiate the system [10]. This consideration is not applicable to NMP-2 since the plant is equipped with a three-train automatically initiated SLC system (as part of the redundant reactivity control system [RRCS]). The RRCS automatically actuates SLC, recirculation pump trip (RPT), alternate rod injection (ARI), and feedwater runback. RRCS actuates automatically after a 98-second delay on high dome pressure or low-low water level. If, after 98 seconds, these signals are still present and sufficient power remains (average power range monitors [APRMs] not downscale or inoperable), RRCS will automatically initiate SLC [9].

The IPEEE report states that the RRCS, recirculation pump trip, and alternate rod injection are not considered because, although the systems are automatic, operator actions are "somewhat more demanding than a transient or small LOCA with SCRAM success," and that the design of the RPS is "fail-safe" and expected to have a high seismic capacity. Since these systems are automatic, operator actions are important only to confirm automatic system operation, and are not the dominant factor in the reliability of the system response. Indeed, the NMP-2 IPE states [9]:

> "The redundant reactivity control system (RRCS) at NMP-2 automatically actuated the standby liquid control (SLC), reactor recirculation pump trip, alternate rod insertion, and feedwater runback. This system was assessed to be reliable and negated the need to model operator actions associated with these functions. Other operator actions associated with level control are not dependent on manual initiation of SLC or the other functions."

If the operator actions were not worthy of modeling in the IPE, they can scarcely be the cause of not considering RRCS in the seismic margin analysis.

RPV Inventory Control

The RPV inventory control function is performed by the RCIC system, or the HPCS system, in the NMP-2 SMA study. No reliance is placed on low-pressure makeup due to the low HCLPF capacity (0.23g PGA) of the nitrogen bottles, which provide the long-term (72 hour) capability to maintain the SRVs open and permit continuation of low-pressure makeup. (The low HCLPF capacity of the nitrogen bottles arises due to seismic interaction with other nearby tanks. Note also that these nitrogen bottles are non-safety related. [9]) The EPRI SMA methodology mandates assuming that offsite power is lost due to the RLE and remains unavailable and unrecoverable for 72 hours [10].

Another issue not identified by the licensee concerning the external nitrogen bottles is the human error probability (HEP) for failing to connect the bottles when needed. The IPE study estimates this HEP to be 0.01 per demand. This necessarily equates the unreliability of all low-pressure injection sources to 0.01 per demand since, if the nitrogen system fails, low-pressure injection fails also (unless there is a stuck-open SRV) [9].

The ability of the SRVs to depressurize the reactor vessel is represented in the IPE (and, apparently, the IPEEE as well) by assuming that all seven ADS SRVs are kept open, when in fact only two are required to maintain depressurization. With all seven valves open, the SRVs remain open for fifteen hours before external nitrogen is supplied. The IPE study reports that the licensee has committed to developing a station blackout procedural modification containing instructions on how to operate the SRVs to minimize depletion of nitrogen (see Section 6.2 of the IPE). Presumably there is no reason why a similar procedure could not be developed for seismic events. Each of the eighteen SRVs (including the seven ADS SRVs) has a nitrogen accumulator that is normally supplied from the gaseous nitrogen storage system via an ADS storage tank, pressure regulator, and the primary containment isolation valves. The nitrogen from the individual accumulator is routed to the solenoids, which port nitrogen to operate the SRV actuator [9].

Under loss of offsite power conditions, and considering a 24-hour mission time (as opposed to the longer 72-hour mission time for the SMA), the internal events IPE for NMP-2 identified the failure probability for HPCS as 0.14 and the failure probability for RCIC at 0.16. The joint failure probability for HPCS and RCIC for a 24-hour mission time under loss of offsite power conditions is the product of these values, or 2.24×10^{-2} per demand. For a 72-hour mission time, these values would increase accordingly for the failure to run term, which would increase the overall failure probabilities above these values. In response to an RAI, the licensee provided values for 72 hours of 0.29, 0.31, and 0.0899. However, the 72-hour values do not include recovery. The licensee states that recovery after 24 hours has a relatively high success probability (presumably due to the length of time available for recovery as a result of lower decay heat levels); however, no quantitative insight into this matter was provided by the licensee. The individual failure rates, as well as the joint failure rate, exceed EPRI NP-6041, Rev. 1, guidelines.

This situation was not identified by the licensee as a vulnerability because the plant Emergency Operating Procedures (EOPs) can be used to mitigate any accident where all structures, systems, and components not on the 0.5g success path are assumed to fail. The EOPs provide direction to the operators to use HPCS and RCIC for inventory control, and to use the RHR system in the suppression pool cooling mode. The licensee is not aware of any EOP or BWR Owners Group Emergency Procedure Guideline (EPG) guidance that must be violated in order to control the plant for 72 hours using only the structures, systems, and components on the 0.5g simplified success path. While not credited in the SMA or the PRA, the nitrogen

system can be used to support low-pressure injection for at least 24 hours following an earthquake. The licensee states that with mitigation for 24 hours, decay heat loads would be sufficiently low to permit time for recovery actions. Finally, there are two success paths which have a HCLPF capacity of 0.23g PGA, which can maintain the plant in safe shutdown for 72 hours. These success paths are not credited in the SMA. The licensee states that the PRA shows the nitrogen system failure to be of limited risk significance.

Accordingly, it does not appear that the ERPI NP-6041, Rev. 1, guidance was followed in this instance. The safety significance of this deviation from the SMA guidance, however, is apparently very low as indicated by the seismic PRA findings and the ability of the plant to maintain shutdown for 72 hours using only structures, systems, and components that are in the 0.5g PGA HCLPF simplified success path. Thus, although the licensee deviated from the EPRI guidelines, the licensee combined a risk evaluation and a timing evaluation to take credit for recovery.

In terms of relevant review findings, the licensee's use of HPCS and RCIC as alternate success paths is contrary to the EPRI NP-6041 guidelines which the licensee adopted for the SMA. The individual HPCS and RCIC failure rates, as well as their combined failure rate, for the high pressure coolant makeup function, all exceed EPRI NP-6041 guidelines. However, the safety significance of this departure from the SMA guidelines is believed to be very low, as discussed above. The licensee could attain enhanced understanding of the seismic severe accident resistance of NMP-2 by considering SLC system as an alternative means for achieving reactivity control following a seismic margin earthquake.

2.1.3 Non-Seismic Failures and Human Actions

The IPEEE submittal states that the identification of success paths and components is based on minimal credit for operator actions (see Section 3.1.2.1 of the IPEEE submittal). Non-seismic failures and human actions are included in the seismic PRA.

The NRC staff provided guidance on consideration of non-seismic failures and human actions in NUREG-1407 as follows (emphasis added):

> "Success paths are chosen based on a screening criterion applied to nonseismic failures and needed human actions. <u>It is important that the failure modes and human actions are clearly identified and have low enough probabilities to not affect the seismic margins evaluation.</u> The screening criteria used in the Maine Yankee margin evaluation (NUREG/CR-4826) addressing both single-train and multi-train systems is an acceptable approach. The redundancies along a given success path should be specifically analyzed and documented when they exist. <u>(In a complementary sense, where a single component is truly "alone" in performing a vital function along a success path, this should be highlighted too.) This information will serve to indicate the extent to which a single failure would or would not invalidate the plant's ability to respond safely to a given earthquake level</u>."

The EPRI SMA methodology does not explicitly address non-seismic failures, however, it recommends that single-train systems with recognized poor availability (e.g., system unavailabilities of more than about 0.01 for the 72-hour success period) "should be treated with caution." Specifically, for the BWR/4 design,

the EPRI NP-6041 states that, for high pressure coolant injection (HPCI) and RCIC, experience has shown that these systems are only moderately reliable, with individual system unavailabilities in the few-percent range. In that case, the EPRI SMA methodology recommends that these two systems be put in series in the success -path logic diagram, so that the combined functional unavailability is sufficiently small [10]. Clearly, this approach should have been followed in the NMP-2 seismic IPEEE for treatment of the HPCS and RCIC systems -- i.e., they should have been combined into a single success path, not represented as alternate success paths. NMPC has, in effect, identified only a single success path for seismic events. However, the seismic PRA results suggests that the safety significance of this departure from the SMA methodology is very small.

Overall, the licensee's consideration of human actions in the SMA was not entirely in keeping with the SMA guidance of EPRI NP-6041, Rev. 1. However, the seismic PRA fully considered human actions and suggests that the safety significance of NMPC's departure from SMA guidance is very low.

2.1.4 Seismic Input

The RLE was defined by the median NUREG/CR-0098 spectral shape anchored to PGA value of 0.5g. Since NMP-2 is a rock site, the peak ground velocity to peak ground acceleration ratio of 36 in/sec/g was used, as recommended in NUREG/CR-0098. According to EPRI [10], this scaling technique is acceptable if the plant is located on a rock site and the two spectra are similar in shape. The licensee states that these conditions are satisfied for NMP-2. According to the licensee: "Both spectra are relatively rich in low frequency power and peak at about the same frequency range."

The licensee's use of the NUREG/CR-0098 median rock spectrum is consistent with the recommendations in NUREG-1407 (Section 3.2.2).

2.1.5 Structural Responses and Component Demands

The seismic review team (SRT) walked down most major components identified in the success-path component list, and determined that a HCLPF capacity of at least 0.5g exists for the seismic margin earthquake (SME) for most components. These walkdown findings were documented on seismic evaluation work sheets (SEWS). In many cases, equipment was inspected and screened out based on SRT knowledge and review. According to the licensee: "Component anchorages were not screened; rather, worst case representative anchorages were selected for analysis to ensure they possessed HCLPFs equal to or higher than the equipment class HCLPF value." (See Section 3.1.4 of the IPEEE submittal.)

Due to equipment configuration, or accessibility limitations, not all components could be screened out during the walkdown. Calculations document the review process and the HCLPF assessments for such components. All these calculations, except one, indicated a HCLPF capacity of 0.5g PGA, or greater. The exception was chatter of HFA Model-154 relay, for which a HCLPF value of 0.45g PGA was calculated. The licensee reports that the calculation is based on the "worst case required response spectra" in the switchgear, and that the acceptance criteria for relay chatter was two milliseconds (ms). The licensee states that associated relay chatter is unlikely to cause an impact at NMP-2. As concluded by the licensee: "For these reasons, and since other conservatisms exist, it is judged that a plant HCLPF of 0.5g or greater exists and this is also assumed in the seismic PRA analysis..."

Structures at NMP-2 were designed in accordance with Regulatory Guide 1.60. The floor response spectra (FRS) for the seismic margin analysis were developed using scaling techniques to adjust the design-basis FRS, as recommended by EPRI [7]. The design-basis earthquake is the Regulatory Guide 1.60 ground response spectrum scaled to a peak ground acceleration of 0.15g. The RLE for NMP-2 is based on the NUREG/CR-0098 median spectral shape scaled to have 0.5 PGA.

GL 88-20 and NUREG-1407 request analysis of bad-actor relays. NMPC went beyond this minimum guidance and evaluated all relays in the success path. With one exception, calculations indicated a HCLPF capacity of at least 0.5g PGA. The exception (as noted previously) is chatter for HFA Model-154 relays, which were calculated to have a HCLPF capacity of 0.45g PGA. This assessment is believed by the licensee to be conservative, because the calculation is based on the worst case required response spectra in the switchgear. In addition, the acceptance criteria for relay chatter was 2 ms, which is stated by the licensee to be unlikely to cause an impact at NMP-2.

The licensee found that all components in the success path screened at a HCLPF value of 0.5g PGA or greater, with the exception of the HFA Model-154 relays. Due to conservatisms in the assessment, the licensee concluded that this component screened out as well.

2.1.6 Screening Criteria

The licensee screened structures, systems, and components, based on whether the HCLPF capacity was estimated to be 0.5g PGA or greater. The licensee's screening criterion is consistent with the SMA's stated scope, and exceeds the NUREG-1407 guidelines for a 0.3g PGA RLE.

2.1.7 Plant Walkdown Process

The plant walkdowns were conducted by the SRT. The SRT members included two members of the plant staff and two consultants from Stevenson & Associates; all SRT members were trained and certified in Seismic Qualification Utility Group (SQUG) procedures. The SRT was supported by five additional individuals (two from NMPC, one from Vectra Technologies, Inc., and two from J.H. Moody Consulting, Inc.). Three members of the latter group participated in nearly all aspects of the IPE and IPEEE development for NMP-2, thus providing coordination between these analyses.

Seismic capability walkdowns were performed by the SRT to achieve the following objectives:

- Screen components that could be shown to have seismic capability above the RLE
- Identify the potential for spatial interactions and other systems interactions (including proximity, II/I issues, seismic spray, and flooding)
- Clearly define failure modes
- Perform preliminary vulnerability assessments

Review walkdowns were also conducted on a case-by-case basis to investigate additional success paths, collect additional information, or verify previous analyses.

Based on the walkdowns, a substantially reduced set of elements remained for detailed review. For each element, it was necessary to perform a demand and capacity evaluation. The demand evaluation considered the level of motion expected at the component and included the amplification of the earthquake motion at upper elevations of the plant. The capacity evaluation consisted of a determination of the ability of the elements to withstand an earthquake. The demand estimates were determined either using a scaling approach or by performing new less-conservative, building response analyses. The demand was compared to the seismic qualification rating. Elements that did not meet comparison limits could have a less conservative demand evaluation performed.

Flooding sources were evaluated and determined by the licensee to be insignificant in terms of seismically initiated flooding in plant buildings. The assessment by the licensee appears to presume that the piping in question is in the original, as-built condition. Clearly, however, there exists a potential for the piping to be in a degraded condition, such that the RLE could fail the piping. The fact that the piping can be degraded is reflected in the failure of such piping causing flooding under circumstances other than a seismic event. For some period prior to the "random" failure, the piping must have been in a degraded condition such that the piping would have failed under RLE loading. This fraction of the time should have been considered in the seismically initiated flooding assessment. This matter could be important for such flooding scenarios as the failure of fire protection water piping at the 261' elevation of the control building, which would result in flooding of the switchgear rooms, and failure of service water piping resulting also in flooding of the switchgear rooms (causing station blackout) [9]. Since such flooding sources were screened out, the seismic PRA also did not consider the conditional probability that the piping could be degraded at the time of a seismic event.

It can be concluded that the licensee's walkdown process was consistent with the guidelines in EPRI NP-6041, Rev. 1, and NUREG-1407.

2.1.8 Evaluation of Outliers

The licensee did not identify any outliers in the seismic portion of the IPEEE. However, as previously explained in connection with the low seismic capacity of the nitrogen system, the licensee failed to follow EPRI NP-6041, Rev. 1 guidance concerning reliance on single-train systems. In order to evaluate the risk significance of this low capacity system, the licensee performed a seismic PRA. The PRA indicates that the safety significance of this deviation from the SMA guidance is very small. Accordingly, the nitrogen system low capacity finding was not identified as an outlier.

Overall, the licensee's consideration of outliers is consistent with NUREG-1407, and the use of seismic PRA to evaluate a potential outlier identified in the SMA appears appropriate.

2.1.9 Relay Chatter Evaluation

EPRI methods were used in the evaluation of relay chatter, along with previous seismic PRAs and the NMP-2 IPE. The relay chatter evaluation was based on a previously developed functional success diagram and the related systems and components that are required to support safe shutdown after an earthquake-induced transient. However, rather than limit its assessment of relay chatter to bad-actor relays, the licensee included all relays in the safe-shutdown path. A total of 181 relays were identified where chatter

could cause failure of a system in the functional success diagram. These relays were subjected to seismic margin screening (at the RLE of 0.5g PGA, rather than the NUREG-1407 value of 0.3g).

A functional relay chatter evaluation was performed to identify those relays to be included in the seismic capability screening and analysis. Table 3.1-3 of the IPEEE submittal summarizes the results of this evaluation. The evaluation considered the expected states of the relays prior to the earthquake and their response to the earthquake. The analysis identified 181 relays where chatter could potentially cause failure of a system in the functional success diagram. Relay chatter impacts on systems and components are summarized as follows:

- The predominant system impact relates to the chatter of protective relays for motor-driven pumps and diesel generators. Chatter results in the tripping of these components, which must be reset locally.

- Chatter of other auxiliary relays associated with actuation of the emergency core cooling system can result in system actuation without a real system demand. In most cases, this occurrence is considered a success for the system. However, the possibility exists for ADS actuation while the diesels are locked out, requiring local reset.

- With the exception of RCIC, relay chatter in valve circuits has a minor impact. A valve can change position or even cycle, but will either reposition itself, or the movement will be small. In the case of RCIC, there are relays that can close the steam supply isolation valves and trip the turbine trip valve. In one case, chatter is recoverable from the control room, and in the other case local recovery is necessary.

The potential for an interfacing-systems LOCA (i.e., LOCA outside containment) as a result of relay chatter was considered and found to be small regardless of seismic margin results. Containment isolation was also reviewed, and it was again determined that relay chatter was not a concern.

In summary, the licensee's relay chatter evaluation was expanded beyond that requested for a focused-scope 0.3g PGA plant. The scope was, however, consistent with the licensee's decision to use a 0.5g PGA RLE for NMP-2.

2.1.10 Soil Failure Analysis

No soil failure analysis was performed since NMP-2 is a rock site, with all safety structures founded on rock. Moreover, soil failure analysis is not considered to be necessary for focused-scope sites [11]. The licensee's approach to soil failure analysis has thus been consistent with NRC staff guidance.

2.1.11 Containment Performance Analysis

The NMP-2 seismic IPEEE containment performance analysis included the following steps:

- Assessment of containment pressure boundary, including the structures, piping, valves, and penetrations.

- Assessment of containment isolation system.

- Walkdown of containment structures, systems, and components, including consideration of penetration configurations and the potential for spatial interactions.

- Assessment of the potential for causing a LOCA outside containment (i.e., an interfacing-systems LOCA)..

- Assessment of the impact of relay chatter on containment isolation.

The primary containment was walked down, including the drywell, vacuum breakers, penetrations (mechanical and electrical), reactor coolant pressure boundary, and instrument lines. Seismic category structures were walked down or, if they were inaccessible, a drawing review was performed. All structures are separated by a three-inch gap to prevent impact, and NMP-2 is a rock site free from soil failure issues.

Overall, the SMA consideration of containment issues has been consistent with applicable guidance. Further, the licensee has quantified containment performance in the supplemental seismic PRA, which indicates that the potential for early containment failure or bypass from seismic events is small.

2.1.12 Seismic-Fire Interaction and Seismically Induced Flood Evaluations

Seismic-fire interactions were discussed in Section 4.8.1 of the IPEEE submittal report. Seismic-fire interactions were addressed using the results of the SMA, seismic walkdowns, fire protection walkdowns, and design review activities. The evaluation considered seismically induced fires, seismic actuation of fire suppression systems, and seismic-induced failure of fire suppression systems.

The seismically induced fire evaluation focused on the potential for seismic events to cause a release of flammable or combustible liquids or gases. Concerning seismic actuation of fire suppression systems, the licensee noted that the fire detection panels at NMP-2 use circuit boards for logic functions, rather than older style panels which were susceptible to inadvertent actuation during earthquakes. Water spray systems with open sprinklers were identified in three buildings for which evaluations were performed. The evaluation of seismic-induced failure of fire suppression systems focused on the II/I issue, noting that NMP-2 was designed and maintained in accordance with II/I requirements.

Fire protection water piping, which is not generally seismically qualified or designed to seismic standards, is mostly rod-hung, welded steel piping. The potential for a fire-water flood was evaluated and considered in the walkdown.

Overall, the licensee's treatment of seismic-fire interactions and seismically induced flooding appears to be consistent with NUREG-1407 guidance.

2.1.13 Treatment of USI A-45

The licensee stated that no weaknesses were identified in the seismic margin analysis with regard to decay heat removal. A plant HCLPF capacity of 0.5g PGA was associated with the RHR system and its support

systems. In addition, using the LLNL seismic hazard results, the licensee calculated the CDF contribution, arising from loss of decay heat removal, at less than 2.2×10^{-7} per reactor year. (It should be noted that containment venting capability is lost when the high-pressure nitrogen system is lost. The seismic PRA indicates that this failure is not significant.)

No weaknesses related to decay heat removal were identified in the seismic IPEEE submittal. Components in the heat removal system were assessed as having a HCLPF capacity of at least 0.5g PGA. Resolution on this basis is consistent with NUREG-1407 guidance.

2.1.14 Other Safety Issues

a. USI A-17

Unanalyzed spatial interactions, as well as interactions due to relay chatter, were considered in the seismic margins study. Control-system interactions that can propagate via the electrical and control systems, due to a seismic event, were also considered.

b. USI A-40

The licensee states in the IPEEE submittal that USI A-40, "Seismic Design Criteria," is not applicable to NMP-2, since the plant was designed to NRC criteria and methods that contain resolution of this issue (which applies to pre-SRP plants).

c. USI A-46

The licensee states in the IPEEE submittal that USI A-46, "Verification of Seismic Adequacy of Equipment," is not applicable to NMP-2, since the plant was designed to NRC criteria and methods that contain resolution of this issue (which applies to pre-SRP plants).

d. Eastern U.S. Seismicity Issue

This issue is considered by the licensee to have been resolved by satisfying the request of GL 88-20, Supplement 4. The LLNL and EPRI seismic hazard results were included in the NMP-2 seismic PRA, and were also used in the development of NUREG-1407 to determine the RLE for the seismic margin analysis of NMP-2.

e. Generic Safety Issues

Some seismic-related information having relevance to Generic Safety Issue (GSI)-172 is provided in the submittal, as discussed in Section 2.4.3 of this TER.

f. Review Findings

The licensee's treatment of the preceding issues is consistent with NUREG-1407 guidance.

2.1.15 Seismic PRA

With the NMP-2 IPE model in hand; with the seismic margins analysis completed, and considering the availability of LLNL and EPRI seismic hazard results for NMP-2, the licensee concluded that the additional effort required to perform a seismic PRA was "comparatively insignificant." The licensee concluded that, since the seismic hazard at the site is low, and the plant HCLPF capacity is high, the risk associated with seismic events would be low. Thus, simplifying conservative assumptions were made to facilitate the analysis.

The following steps were taken in performing the seismic PRA for NMP-2:

- Both the LLNL and EPRI seismic hazard curves were used in quantifying the seismic PRA model.
- Seismic fragilities were developed from the seismic-margin HCLPF values with one exception -- a generic loss of offsite power fragility from Seabrook Station was used.
- The IPE analysis (1992) was used to estimate the conditional probability of loss of offsite power given an earthquake initiating event.
- Non-seismic failure rates from the NMP-2 IPE study were used directly.
- An event-tree model was developed from the IPE to generate and quantify seismic accident sequences.
- The model was quantified using the PLG, Inc., RISKMAN computer code, which was also used for the IPE.

A major assumption made in the seismic PRA is that ground motions exceeding the plant-level capacity (characterized by a surrogate element having a HCLPF value of 0.5g PGA) are modeled to result directly in core damage. Such a scenario is assumed to lead to an unisolated containment, with loss of injection, at a frequency of 1.6×10^{-7} per reactor year (EPRI) to 9.0×10^{-7} per reactor year (LLNL). This accounts for about 65% (EPRI) and 75% (LLNL) of the total seismically induced CDF. Failure of the non-seismic qualified high-pressure nitrogen accumulators (HCLPF of 0.23g PGA) was assumed in the seismic PRA to result in failure of low-pressure injection in the long term. The CDF associated with this failure mode was assessed at 3.8×10^{-8} per reactor year (EPRI) to 1.6×10^{-7} per reactor year (LLNL). [This scenario is assumed to result in an initiating event, even if there is no loss of offsite power. Losing all nitrogen and instrument air results in loss of feedwater and loss of the main condenser; it also results in loss of low-pressure injection if the operators have not placed the plant on shutdown cooling in time.] The licensee considers this scenario to be conservative "because the operators are likely to be on shutdown cooling before nitrogen is needed." The frequency of core damage due to station blackout (seismically initiated loss of offsite power, together with non-seismic failure of the diesel generators) was assessed at 3.4×10^{-8} per reactor year (EPRI) to 1.0×10^{-7} per reactor year (LLNL). This result is also considered to be conservative since success of RCIC or HPCS could extend the timing, and increase the chance, of recovery, whereas the sequence was modeled as an early core-damage sequence.

To convert the HCLPF results from the seismic margin study to median fragilities, the licensee multiplied the HCLPF value by a factor of 2.13, based on its assessment of guidance in an EPRI report [10]. Using the simplified seismic PRA approach, the licensee calculated CDF results of 2.5×10^{-7} per reactor year using the EPRI hazard curve and 1.2×10^{-6} per reactor year using the LLNL hazard curve. The licensee considers these results to be conservative, due to modeling assumptions and a conservative conversion from seismic margin HCLPF values to seismic PRA fragilities.

The licensee has thus gone beyond NUREG-1407 guidance for a seismic margin analysis, and extended the study to the level of a seismic PRA, in order to provide risk management insights.

2.1.16 Process to Identify, Eliminate, or Reduce Vulnerabilities

No specific criteria to identify vulnerabilities were identified in the IPEEE submittal. The licensee relied on the IPEEE process, the conformance of the plant to the conservatively established 0.5g RLE, and the low seismic CDF results (particularly compared with the internal events IPE results) to conclude that no vulnerabilities exist with respect to seismic initiators.

The IPEEE process used by the licensee is capable of identifying seismically related severe accident vulnerabilities. No vulnerabilities were found.

2.1.17 Peer Review Process

The licensee employed an internal peer review within the IPEEE team, as well as an independent in-house peer review performed by NMPC personnel not involved in conducting the IPEEE study itself. Additionally, the licensee requested the New York Power Authority to review a draft of the IPEEE.

The licensee's peer review process is apparently consistent with NUREG-1407 guidance.

2.2 Fire

A summary of the licensee's fire IPEEE process has been described in Section 1.2. Here, the licensee's fire evaluation is described in detail, and discussion is provided regarding significant observations encountered in the present review.

2.2.1 Overview and Relevance of the Fire IPEEE Process

a. Methodology Selected for the Fire IPEEE

The analysis of internal fire risk utilizes both the EPRI FIVE methodology [12] and fire PRA methods. The following summarizes the approach and methods used in this analysis:

1. Utilizing the FIVE methodology, compartment boundaries were evaluated, and fire-ignition frequencies were developed for each compartment. Also, Appendix R exemptions and deviations were assessed to ensure that their potential impacts on the IPEEE analysis were understood. A plant walkdown was included as part of this analysis.

2. A computerized spatial database was developed so that all plant cables and components in a fire zone could be identified. This was necessary to identify the impacts of a fire on systems and components in each area.

3. Location dependencies were identified for the offsite power supplies, main feedwater, main condenser, and their support systems. This treatment provides additional success paths and results in improved plant reliability for screening and evaluation areas. The IPE was used to identify the systems and dependencies necessary to support these key functions. Then, cable block diagrams were developed, identifying critical cables. With these cables and their impact on the IPE identified, the spatial database was utilized to determine the fire zones where these critical cables were located.

4. The spatial database, the Appendix R database, and the location dependencies for non-Appendix R equipment were all used to identify component and system impacts on the IPE, due to a fire in each area. Initial screening assumes the fire fails all cables and components in the area. Fire impacts include consideration of initiating events (plant trip or immediate shutdown) and unavailability of systems modeled in the IPE.

5. Based on the impact and frequency of a fire in the area, a screening process was used to determine whether a fire in the area represented an insignificant contribution to CDF or whether a detailed analysis needed to be performed. The IPE is used to support both quantitative and qualitative screening judgements. This task was equivalent to accomplishing the FIVE qualitative and conservative quantitative screening.

6. Those areas that did not screen out during the initial screening analysis (Item 5 above) were evaluated in greater detail to establish realistic scenario frequencies, or to screen the areas out. This analysis considered each unscreened area in greater detail, including considerations of proximity of important cables, fire severity, fire causes, and suppression. At this point in the analysis, fire modeling aspects of FIVE (i.e., identifying targets and sources, combustible loadings, damage thresholds, and suppression) were used, as necessary, to support the evaluation. Plant walkdowns were an important part of the detailed analysis strategy for screening areas.

7. Containment performance, fire risk scoping study issues, and USIs were assessed with regard to impact on public safety.

b. Key Assumptions Used in Performing the Fire IPEEE

No list of assumptions is specifically provided in Reference [1]. However, some of the key assumptions, which are noted in the text of Reference [1], include:

1. Ten fire panel fires were not considered because there were no reported fire panel fires at NMP-2.

2. The heat release rate for an electrical cabinet fire was assumed to be 65 Btu/sec.

3. The FIVE modeling methodology is appropriate for screening purposes only, and is not capable of evaluating scenarios involving intervening combustibles and of modeling fire growth.

4. No explicit credit is taken for manual suppression.

5. A plant-specific analysis to estimate automatic fire protection system unreliability was not deemed necessary.

6. All cables at NMP-2 are IPEEE 383 qualified. Therefore, self-ignited cable fires are not considered.

7. The only seismically induced fire sources considered in the IPEEE were releases of flammable or combustible liquids or gases.

8. Inadequate operation of a carbon dioxide system will not result in any equipment operability concerns.

9. Inadvertent operation of a water deluge system in the RCIC room will not release enough water to damage RCIC components.

10. Fire protection systems have been installed in accordance with National Fire Protection Association (NFPA) codes and standards. Therefore, adequate assurance is provided that fire protection systems will not fall on safe-shutdown components during a seismic event.

11. Based on a review of available technical information relating to smoke damage, there is no concern for operability of safe shutdown equipment outside the area of fire origin.

12. Since wet pipe systems, dry pipe systems, and preaction systems each require the operation of individual sprinklers to cause system water flow, these systems are not susceptible to water discharge due to seismically induced actuation.

c. Status of Appendix R modifications

The status of Appendix R modifications was not provided in the IPEEE submittal.

d. New or Existing PRA

The IPEEE is a new study which combines aspects FIVE methodology and fire PRA.

2.2.2 Review of Plant Information and Walkdown

a. Walkdown Team Composition

Several walkdowns were performed in support of the fire IPEEE analysis. An initial walkdown was performed to investigate fire barriers, presence of ignition sources, and issues associated with the Sandia Fire Risk Scoping Study. Other walkdowns were performed in support of the screening and detailed analysis efforts. The initial walkdown was conducted by Robert C. Beller, P.E., Senior Fire Protection Engineer (Pacific Nuclear) and Gaines Bruce (NMPC Fire Protection Engineer).

b. Significant Walkdown Findings

The following summarizes the insights gained from the plant fire walkdowns:

1. For the service water pump area, the following findings were noted in Reference [1]:

   - The pumps are in a deep pit, about forty feet below the deck level (El 261), where most of the remaining service water equipment (strainers, motor-operated valves [MOVs], motor control center [MCC], cable trays) are located. There is another elevation above the deck where hot gases would tend to collect given a fire in the area. There are no important equipment or cables at this higher elevation.

   - There is sufficient distance between the three pumps, their associated cables, and other critical equipment on the upper deck, such that a fire initiated at one pump is very unlikely to impact a second pump (let alone all three as assumed in the screening analysis). The only conceivable scenario that could impact more than one pump might be a pump fire and oil spill onto the floor. However, there is limited oil associated with a pump motor, and there is a large surface area associated with oil spread on the floor, minimizing potential impact to all three pumps. This event may not be credible, but if it were, the frequency of such an event is less than assumed in the screening analysis. In addition, the safety-related header supplies are not impacted. Our judgment is that pump fires that impact all three pumps are unlikely, and they can be screened out (based on CDF less than 10^{-7} per year).

   - There are two large unit coolers (e.g., 2HVY*UC2A & C in FA61) below the deck at El 261, but above one of the pumps. The initiating frequency for a unit cooler fire is less than the frequency for the pumps, and the impact would likely be for one pump only. Therefore, this source can be screened out as stated above.

   - The major electrical cabinet in each area is the MCC (e.g., 2EHS*MCC101 in FA61) on the deck at El 261. The MCC is actually contained within another cabinet, thus the impact of fires within the MCC are very unlikely to impact cable trays above the MCC. No vented cabinets that could impact cables or conduits were identified during the walkdown.

   - Each MCC supplies divisional strainers, strainer MOVs, pump discharge MOVs, unit coolers, a crosstie MOV, header MOVs, and tunnel heaters. There are no cables that would de-energize the pumps and, since service water is normally operating (MOVs are in their correct position for normal operation), a fire would have to cause a short circuit to have an impact on system operation (i.e., shorts cause pump discharge MOV or header MOV to close).

2. For the normal switchgear rooms, the following findings were noted in Reference [1]:

   - <u>FA51</u> - This fire area contains the chargers and switchgear that are associated with the 125 VDC systems 2BYS-SWG001A and 2BYS-SWG001B. The power cables for 2NNS-

SWG016 are routed through this area. There are numerous cables associated with the 115 kV motorized disconnects. A fire in this area results in the loss of offsite power to 2ENS*SWG101 due to cables; cabinet fires do not fail offsite power supplies.

- EA52 - A fire in this area can result in a total loss of offsite power due to control cables; cabinet fires do not fail offsite power supplies. Power cables to both 2NNS-SWG016 and 017 are routed in this area, there is approximately 50 feet separation between the cables (cables to SWG016 are routed on the opposite end of the room from SWG017 cables).

- EA78 - A fire in this area can result in a total loss of offsite power due to control cables; cabinet fires can cause a partial loss of offsite power (loss of 2NNS-SWG016 supply).

- EA79 - A fire in this area can result in a total loss of offsite power due to control cables; cabinet fires can cause a partial loss of offsite power (loss of 2NNS-SWG017 supply).

c. *Significant Plant Features*

No significant plant features were noted in Reference [1].

2.2.3 Fire-Induced Initiating Events

A variety of potential fire-initiated transient events was considered in the IPEEE analysis. All fires except control room fires were screened out. For each control room fire, the resulting initiating events could be verified.

a. *Were Initiating Events Other than Reactor Trip Considered?*

The fire initiating event sequences which were modeled consist of:

- General plant transient
- Loss of offsite power
- Station blackout
- Stuck-open SRV
- Total loss of service water

b. *Were the Initiating Events Analyzed Properly?*

Generally, a fire scenario would result in one of the foregoing event sequences, depending upon the location of the fire and the equipment affected. However, depending on random equipment failures, the scenario may propagate into a consequential transient.

For each scenario described in the submittal (which all involve the control room), a listing of damaged cables is provided. Therefore, the initiating events which result from fire damage could be verified. A verification was performed for a few fire scenarios. For these fire scenarios, the initiating events were analyzed properly.

2.2.4 Screening of Fire Zones

a. Was a Proper Screening Methodology Employed?

The following summarizes the basic approach used in the IPEEE to screen fire areas:

- A computerized spatial database was developed, so that all plant cables and components in a fire zone could be identified.

- Location dependencies were identified for the offsite power supplies, main feedwater, main condenser, and their support systems. This treatment provides additional success paths, and results in improved plant reliability for screening a number of areas. The IPE was used to identify the systems and dependencies necessary to support these key functions. Then, cable-block diagrams were developed, identifying critical cables. With the cables and their impact on the IPE identified, the spatial database was queried to determine the fire zones where these critical cables are located.

- The spatial database, the Appendix R database, and the location dependencies for non-Appendix R equipment were used to identify component and system impacts, given a fire in the zone. Initially, the fire was assumed to fail all cables and components in the zone. Fire impacts include consideration of initiating events (plant trip or immediate shutdown) and unavailability of systems modeled in the IPE.

- Based on the impact and frequency of a fire in the area, a screening process was used to determine whether a fire in the area represents an insignificant contribution to public safety, or whether additional more-detailed analysis should be considered. The frequency of a fire in the area was based on the FIVE evaluation. The IPE was used to support both qualitative screening judgements and quantitative screening.

Several screening techniques were utilized in this evaluation, as summarized below:

- Quantitative screening using the IPE was performed for several fire zones. An initiating-event fire for a specific fire zone was defined, and event tree rules were revised to account for the fire impact. If the annual CDF was less than 10^{-6}, the zone was screened out.

  This screening criteria is considered reasonable because the impact of a fire is conservatively assumed to fail everything in the zone. Typical reduction factors (i.e., geometric and severity factors within an area, given a fire in the area) in fire PRAs are on the order of 0.1, or less. Thus, the CDF should be on the order of 10^{-7} or less, which is less than 1% of the IPE CDF.

- If a fire does not cause an initiating event, the unavailability of systems in the fire zone, based on the fire frequency, are compared to the IPE. In general, the unavailability from a fire (e.g., frequency of a fire, taken over a 24-hour mission time) is small in comparison to unavailabilities from the IPE. In cases where there was significant damage to safety related equipment, an initiating event was assumed in this initial screening analysis.

- Qualitative screening was used when no initiating event or significant impact on IPE systems could be identified (i.e., when it was obvious that risk quantification would result in values less than 10^{-7} per year).

b. Have the Cable Spreading Room and the Control Room Been Screened Out?

The cable spreading room has been screened out. The screening of the cable spreading room is deemed to be proper due to: (1) the existence of only cables in the spreading room, and (2) the existence of an automatic fire suppression system in the cable spreading room.

c. Were There Any Fire Zones/Areas that Have Been Improperly Screened Out?

There are no fire areas which were found to be improperly screened out. However, without either plant layout drawings or a plant walkdown, this review could not determine whether or not adequate consideration was given to cross-zone fire and smoke spread, or whether or not there may be fire suppression activities which could result in safety-related equipment damage in adjacent plant areas. The only fire area which survived screening was the control room.

2.2.5 Fire Hazard Analysis

The development of NMP-2 fire ignition frequencies was based upon nuclear industry data assembled in the EPRI fire events data base [13] and the area loads with fire sources and combustibles (i.e., the method recommended in the EPRI fire events data base). Plant-specific fire frequency data was used only to develop the industry-wide generic fire frequencies.

2.2.6 Fire Growth and Propagation

a. Treatment of Cross-Zone Fire Spread and Associated Major Assumptions

The NMP-2 analysis does not provide any information regarding the treatment of cross-zone fire spread, nor of any associated major assumptions.

b. Assumptions Associated With Detection and Suppression

The following are the assumptions associated with detection and suppression of fires:

- Implicit credit for manual suppression of fires was taken only for control room fire scenarios; and

- Automatic detection and suppression was credited in the screening analysis, with an unreliability of 0.05 per demand.

c. Treatment of Suppression-Induced Damage to Equipment, if Available

The following plant areas were analyzed for suppression-induced damage to equipment:

- <u>Wet Pipe Systems</u>

- radwaste building
- reactor building cable tunnels
- control building
- diesel fire pump room
- turbine building

- Dry Pipe Systems

  - turbine building
  - standby gas treatment building

- Preaction Systems

  - reactor building
  - emergency diesel generator rooms
  - turbine building

- Water Spray Systems

  - turbine building
  - RCIC pump room
  - radwaste building

Inadvertent operation of carbon dioxide fire protection systems was assumed not to result in any equipment operability concerns.

2.2.7 Evaluation of Component Fragilities and Failure Modes

a. Definition of Fire-Induced Failures

Fire-induced failures were defined as loss of function of equipment associated with damaged cables, damaged MCCs, or damaged equipment itself.

b. Method Used to Determine Component Capacities

Components were assumed to either fail with a probability of 1.0 (if the fire was sufficiently close) or they had a chance of success based on reliability and availability models in the IPE (IPE also identifies failure modes). The following summarizes the treatment of component failures:

- In the initial screening analysis, all equipment in the compartment being analyzed were assumed to fail. There was no credit taken for detection or suppression.

- In the detailed analysis, equipment that are fire sources were assumed to fail, and target equipment were assumed to fail in the zone of influence (i.e., plume and hot gases). No explicit credit for manual suppression was taken in the analysis.

c. Fragilities

Generic fragilities have been utilized for cabling in the fire-related damage evaluation.

e. Technique Used to Treat Operator Recovery Actions

It is unclear from the information provided in Reference [1] what technique was employed to treat operator recovery actions. Also, the operator recovery probabilities for the control room fire scenarios appear to be highly optimistic. No details are provided concerning the methodology employed to calculate the likelihood that the recovery action is unsuccessful (e.g., THERP, SHARP, HCR). For fire-initiated sequences, there are performance shaping factor (PSF) issues, which are unique to fire situations, and would not have to be assessed in the IPE human reliability analysis. These PSF issues mostly relate to environmental stressors (e.g., the impact of smoke and suppression agents, reduced visibility, impaired communications due to the use of breathing apparatus) and psychological stressors (i.e., the occurrence of an unexpected event such as fire of sufficient severity to cause equipment failures). The operator failure rates for the fire sequences in the NMP-2 IPEEE submittal report (pages 4.6-58 to 4.6-62) are assigned some relatively low values (ranging from 10^{-3} to 10^{-1} per demand) considering the factors leading to higher levels of stress. No indication is provided in the IPEEE submittal report that these factors were considered, nor is an indication provided of how much time is available to perform the human recovery actions.

2.2.8 Fire Detection and Suppression

The automatic suppression systems at NMP-2 consist of water, carbon dioxide, and Halon-based systems. The unavailabilities utilized for these systems are consistent with probabilities reported in past fire PRAs.

Manual fire suppression was credited implicitly for control-room fire scenarios, but not credited for all other fire areas.

2.2.9 Analysis of Plant Systems and Sequences

a. Key Assumptions Including Success Criteria and Associated Bases

The success criteria were taken directly from the IPE.

b. Event Trees (Functional or Systemic)

Only a limited number of the internal event trees were determined by the licensee to be applicable/relevant to fire CDF quantification. These include:

- general plant transient
- loss of offsite power
- station blackout
- stuck-open SRV
- total loss of service water

These applicable fire sequences are typical of those reported in past fire PRAs.

c. Dependency Matrix, if it is Different From That for Seismic Event

No dependency matrix was provided.

d. Plant-Unique System Dependencies

It is unknown if there are any plant-unique system dependencies.

e. Most Significant Human Actions

The only recovery action listed in Reference [1] is stabilization and recovery of the plant, by maintaining RPV level and heat removal from the control room or from the remote shutdown panel.

2.2.10 Fire Scenarios and Core Damage Frequency Evaluation

The licensee has properly demonstrated and summarized how the CDF was estimated for each fire scenario.

2.2.11 Analysis of Containment Performance

a. Significant Containment Performance Insights

The following containment failure modes were identified and evaluated by the licensee:

- Containment isolation/bypass
- Containment overpressure failure

The licensee concluded that fires were insignificant contributors to the above containment failure modes.

b. Plant-Unique Phenomenology Considered

No plant-unique phenomenology was considered.

2.2.12 Treatment of Fire Risk Scoping Study Issues

Six issues were identified in the Fire Risk Scoping Study [4]. Control systems interactions were addressed in the context of control room fires, with the licensee concluding that all controls required for safe shutdown have transfer or isolation switches located outside the control room. Seismic/fire interactions were addressed by focusing on the potential for seismic events to cause a release of flammable or combustible liquids or gases, by evaluating the potential for seismic actuation of fire suppression systems, and by examining the potential for seismic-induced failure of fire suppression systems. Weakly anchored electrical cabinets have been found to be an important seismically induced fire risk contributor [14]; however, these have not been considered. Manual fire fighting effectiveness (including smoke control) was addressed by comparing the fire brigade and fire protection attributes of NMP-2 against the EPRI

evaluation of the Fire Risk Scoping Study. The licensee concluded that NMP-2 meets all the attributes identified by EPRI. The potential adverse effects on plant equipment by combustion products (including smoke) were addressed only with respect to short-term response to the fire; the operators were assumed to be able to shut down the plant without experiencing additional equipment losses due to smoke. Total environment equipment survivability (including spurious operation of suppression systems) was addressed by considering the potential short-term adverse effects of combustion products on plant equipment (for the long-term, the IPEEE assumed that the operators would be able to shut down the plant without experiencing additional equipment losses due to smoke damage). Short-term damage due to smoke was assessed as being mitigated by plant-specific design features. Spurious or inadvertent fire suppression system actuation was also assessed as being mitigated by design features; consideration of a specific architect-engineer review of historical events identified in NRC Information Notice 83-41 was also cited. The issue regarding the adequacy of fire barriers was addressed only qualitatively; no fire-barrier failure rates were used in the analysis. The issue regarding the adequacy of analytical tools for fire assessments was addressed by using the FIVE methodology, which had been previously approved by the NRC for fire IPEEE assessments.

a. Assumptions Used to Address Fire Risk Scoping Study Issues

1. A plant-specific analysis to estimate automatic fire protection system unreliability was not deemed necessary.

2. The only seismically induced fire sources considered were releases of flammable or combustible liquids or gases.

3. Inadvertent operation of a carbon dioxide system would not result in any equipment operability concerns.

4. Inadvertent operation of a water deluge system in the RCIC room will not release enough water to damage RCIC components.

5. Fire protection systems have been installed in accordance with NFPA codes and standards. Therefore, adequate assurance is provided that fire protection systems will not fail on safe-shutdown components during a seismic event.

6. Based on a review of available technical information relating to smoke damage, there is not a concern for operability of safe-shutdown equipment outside the area of fire origin.

7. Since wet pipe systems, dry pipe systems, and preaction systems each require the operation of individual sprinklers to cause system water flow, these systems are not susceptible to water discharge due to seismically induced actuation.

b. Significant Findings

1. Fire barrier failures were not analyzed.

2. Potential adverse effects on plant equipment by combustion products were not addressed.

3. The fire brigade and fire protection attributes were checked against the EPRI evaluation for the Fire Risk Scoping Study. The licensee concluded that the fire protection program meets all attributes listed by EPRI.

4. All controls for systems required to achieve and maintain safe shutdown, in the event of a fire within the control room, have transfer or isolation switches located outside the control room. Procedures are in place which outline the shutdown strategy, utilizing the remote shutdown system, and actions to be taken prior to evacuating the control room in the event of a fire. Therefore, the licensee concluded that the issue of control system interactions had been adequately addressed.

2.2.13 USI A-45 Issue

a. Methods of Removing Decay Heat

The HPCS, RCIC, and RHR systems are the methods considered for decay heat removal during and after a fire event.

b. Presence of Thermo-Lag

Thermo-Lag is not used at NMP-2.

2.3 HFO Events

The licensee used a progressive screening approach based on Section 5 of NUREG-1407 [3] to assess HFO events (high winds, tornadoes, external floods, transportation and nearby facility accidents, and other plant unique external initiators). The licensee reviewed the plant for conformance with the SRP and screened all HFO events on the basis of conformance to the SRP criteria.

2.3.1 High Winds and Tornadoes.

2.3.1.1 General Methodology

The licensee screened out high winds and tornadoes based on compliance with the 1975 SRP.

2.3.1.2 Plant-Specific Hazard Data and Licensing Basis

The IPEEE submittal reports that between 1951 and 1980, there were 14 tornadoes in the 14,000 square mile area surrounding the plant site. Two tornadoes occurred within 5.6 miles of the plant. Based on statistical analysis of these events, the licensee's Updated Safety Analysis Report (USAR) for the plant calculates a tornado strike frequency of 3.57×10^{-5} per year. The plant is designed for a 360 mph wind velocity (considering the sum of the rotational velocity of 290 mph and the translational velocity of 70 mph).

2.3.1.3 Significant Changes Since Issuance of the Operating License

The submittal did not identify any significant changes since issuance of the operating license.

2.3.1.4 Significant Findings and Plant-Unique Features

No significant findings were cited in the submittal.

2.3.1.5 Hazard Frequency

Notwithstanding the conformance of the design to 1975 SRP criteria, the licensee examined two tornado-induced core damage scenarios. First, the licensee considered the possibility that a large missile might dislodge from non-safety related buildings and damage plant systems. The licensee quantified this frequency at 3.6×10^{-8} per year, based on the following:

- Tornado strike frequency of 3.57×10^{-5} per year
- Conditional probability of significant missile generation equal to 0.5
- Conditional probability of 0.1 that the missile hits a safety-related structure and pierces the shielding
- Conditional core-damage probability of 0.01, given the missile strike; also, a conditional probability of failing equipment whose functionally redundant equipment subsequently fails, also of 0.01; for a total conditional core-damage probability contribution of 0.02.
- Tornado-induced CDF due to missile impact is 3.6×10^{-8} per reactor year.

This analysis ignores the possibility that a missile strikes the diesel generator building and takes out one diesel. The conditional probability of the second diesel failing is not 0.01, but is rather closer to 0.1, particularly for the long term. Additionally, loss of offsite power may be assumed for a tornado strike at the facility due to the vulnerability of the switchyard to damage. Thus, a conservative assessment of the frequency of station blackout, resulting from a tornado strike, would be made as follows:

- Tornado strike frequency of 3.57×10^{-5} per year
- Conditional probability of significant missile generation equal to 0.5
- Conditional probability of 0.1 that the missile strikes the diesel generator building and pierces the shielding, disabling a diesel generator
- Conditional probability of 0.1 for failure of the second diesel due to random causes (considering failure to start, failure to run, and maintenance contributions)
- Tornado-induced CDF due to station blackout is approximately 2×10^{-7} per reactor year.

Even when quantified on this crude basis, the CDF is a factor of five below the IPEEE screening criterion of 10^{-6} per reactor year.

For the second tornado-induced core-damage scenario, the licensee examined the frequency of a Nine Mile Point, Unit 1, stack failure damaging the hydrogen storage building, resulting in an explosion. The licensee quantified this frequency as being 5.9×10^{-8} per reactor year, based on the following approach:

- Tornado strike frequency of 3.57×10^{-5} per year

- Conditional probability of 0.5 for failure of the Nine Mile Point, Unit 1, stack

- Conditional probability of 0.03 that, when the stack fails, it strikes the hydrogen storage facility and causes an explosion (derived from the fraction of the 360° radius around the stack occupied by the hydrogen storage facility, and assuming a probability of unity for an explosion if the facility is struck by the stack)

- Conditional probability of 0.1 for core damage, given an explosion; also, a conditional probability of failing equipment whose functionally redundant equipment subsequently fails, of 0.01; for a total conditional core-damage probability contribution of 0.11.

- Tornado-induced CDF due to Unit-1 stack collapse is 5.9×10^{-8} per reactor year.

Summing these results, the total tornado-induced CDF is estimated at 2.6×10^{-7} per reactor year. This value would result in screening-out of tornados based on NUREG-1407 guidance for a bounding PRA. Hence, the licensee correctly screened tornadoes and high winds as an insignificant contributor to external events risk.

2.3.2 External Flooding

2.3.2.1 General Methodology

As previously noted, the licensee screened external flooding based on SRP conformance. The submittal states that a re-evaluation of maximum precipitation (pertaining to GI-103) was reported in the NMP-2 USAR. No details of this re-evaluation are presented in the submittal. The licensee included an extensive discussion of external flooding, attempting to make bounding arguments to the effect that external flooding is an insignificant contributor.

2.3.2.2 Plant-Specific Hazard Data and Licensing Basis

While noting the screening of this event based on SRP conformance (as provided for in NUREG-1407), this review finds that the licensee's bounding arguments appear to be flawed. Without be-laboring the point, the following comments are offered to illustrate the problems with the licensee's bounding assessment.

The licensee's flooding analysis addressed the potential for lake flooding, overland flooding, and heavy precipitation to damage critical plant equipment and/or structures. These effects included water entering

from the outside, as well as heavy roof loads. Flooding due to plant internal sources (such as tank ruptures) was previously evaluated in the IPE study.

The NRC safety evaluation report (SER) on NMP-2 identified the potential for flooding at El 262.5 ft. The IPEEE submittal states:

> "Should water level reach 262.5' in the Control Building, the three emergency switchgears will almost certainly fail. These failures would result in a very high, i.e., approaching 1.0, conditional core damage probability. Thus, the probability of water reaching 262.5' in the Control Building could be important to plant risk."

The licensee evaluated historical data, and concluded that it could not be shown that the CDF would be below 10^{-6} per year (probable maximum flood [PMF] frequency of 10^{-5} and a conditional core damage probability of 0.1 or less). Rather, the licensee noted that the PMF would develop over a period of time, which would make possible the implementation of mitigation measures.

The following comments regarding this assessment are in order:

- There is some validity to the position that the PMF would evolve over a large amount of time. However, it is not uniformly true that a long lead time would be available. Moreover, even when a long lead time is available, it will not always be the case that it will be used to an advantage. That is, there is a potential for errors in judgment that would result in a flood-related core damage accident even though, in retrospect, there was sufficient time to avoid it. To accomplish a reliable evaluation of these considerations, it is necessary to construct an event tree showing the various equipment needed for safe shutdown, possible operator prevention and recovery actions (and their timing), and so on [14]. Timing considerations must also be reflected in the assessment (e.g., range of lead times, range of times from scram to cold shutdown, range of times from cold shutdown to loss of decay heat removal, range of times from loss of decay heat removal to core uncovery and core damage).

  The IPEEE submittal states that "ample time would be available for plant operators to place the plant in a safe condition and perform recovery actions such as sandbagging the three control room doors, caulking outside and inside Control Building doors, installing pumps, and possibly reconfiguring plant electrical components." While all of this reasoning may be valid, the potential for such actions to recover the situation, or to prevent the flooding in the first instance, is not unbounded. Some notion of timing and likelihood is necessary in order to place these actions into perspective.

- Frequencies for severe weather phenomena, including flooding, are difficult to estimate, particularly as one goes beyond the historical data to periods of hundreds to hundreds of thousands of years. As one source has noted: "... even if one could examine accurate weather data for the past 100,000 years, there would still be significant uncertainty as to whether the probabilities developed from that data would be truly applicable to the next fifty or so years" [15].

- It is not clear that the licensee's assessment considered all sources of flooding. For lake sites, flooding could occur due to combinations of high-lake water level, wave effects, high wind-driven

water levels (including wave runup and wind setup), surges, seiches, ice jams, etc. [15, 16, 17]. Wind-related waves would seem to be a potentially important factor which the licensee's analysis does not explicitly consider.

- Adequate consideration of flooding should not be limited to examination of individual sources alone – i.e., combinations are possible, particularly when one is considering flood frequencies in the range from 10^{-4} to 10^{-5} per year or less. Some combinations can be dismissed easily, but other combinations are much more likely, and the rarest flooding events undoubtedly involve combinations of extreme phenomena or extreme conditions. Reference [14] cites concerns specifically for plants on the Great Lakes "for which the problem arises due to the possible (rare) combination of several effects such as storm-driven wave runup, wind-generated waves, and an unusually high lake level."

- In the systems analysis response to flooding, the analyst must take into account such factors as random (non-flooding) unavailability of equipment (due to surveillance testing, maintenance, human error, or random failure). Potentially correlated factors must also be considered (such as the potential for long-term loss of offsite power, inability to deliver diesel fuel to the site, to support long-term diesel operation, etc.).

- Specifically regarding nuclear power plant sites on the Great Lakes, Reference [14] notes:

> "Of course, lake levels rise and fall over the years, for a variety of reasons both natural and man-made. For the Great Lakes, only slightly more than 100 years' data exist. While extrapolations out to a few hundred years are routinely done for planning purposes, it is difficult to know how reliable these are, especially in the light of the rise in Great Lake levels over the past decade or so that is not well explained ...
>
> Effects of extreme winds, including both wind-driven waves and wind setup along the shore, are often much larger than the variations in the lake levels themselves: for example, Lake Michigan data cited by Kimura and Budnitz ... show only about two feet difference between the 10-year (known) and 50-year (extrapolated) lake levels in comparison to 5-foot or even up to 10-foot effects from wind and wave phenomena at certain sites.
>
> Analysis of a given site requires knowing the subsurface topography and local configuration. Theoretical understanding of wind-wave effects is reasonably well grounded, and reliable for modest extrapolations beyond the historical record.
>
> Evaluation: The historical record can support F_F values down to the range of about 0.01 per year. Extrapolations to another order of magnitude, to the range of about 0.001 per year, can be made with modest confidence. Beyond that, uncertainties become so great that it would be difficult to rely heavily on analysis using such extrapolations."

2.3.2.3 Significant Changes Since Issuance of the Operating License

The submittal did not identify any significant changes since issuance of the operating license.

2.3.2.4 Significant Findings and Plant-Unique Features

No significant findings were cited in the submittal.

2.3.2.5 Hazard Frequency

Defining external flood frequencies down to the range of 10^{-5} per year involves large uncertainties. Equally as clear, there are known potential sources of flooding at that frequency (or greater) which could produce extremely large floods. Some balance needs to be drawn between this information and the amount of lead time available to avoid core damage in the event of extremely large floods.

2.3.3 Transportation and Nearby Facility Accidents

2.3.3.1 General Methodology

The analysis for transportation and nearby facility accidents utilized compliance with the 1975 SRP as a basis for screening all potential hazards.

2.3.3.2 Plant-Specific Hazard Data and Licensing Basis

Only one manufacturing or industrial plant (Alcan Aluminum Corporation's Alcan Sheet and Plate Division) is located within 8 km of NMP-2. There are no chemical plants, refineries, military bases, or underground gas storage facilities within 8 km of the plant. There are no pipelines within 8 km of the plant, except the Sithe Energies facility, discussed below. The principal roadway within proximity of NMP-2 is Route 104, which passes 6.2 km to the south of the plant. Highway access to the site is via two county routes, Route 1A to the southwest and Route 29 to the east. A private east-west roadway crosses the site and connects these two county routes.

One railroad company, Conrail, transports freight in the vicinity of the plant. The closest rail line to the site is the Oswego-Mexico branch of Conrail, located 2.5 km from the site. This line has daily service on demand, and averages one train daily, five days a week. A rail spur was constructed to serve NMP-2 during construction and operation. No explosive or flammable material is transported on this route. The licensee reports that the distance to the rail line exceeds the safe distance for truck traffic, as specified in Regulatory Guide 1.91.

The Oswego River passes within 11 km of the site and serves as the major route for waterborne traffic on Lake Ontario. Ships passing in commercial lanes pass no closer than 11.3 km from the intake structures of NMP-2. Since this distance exceeds the 10 km distance, potential explosions on a ship or barge are not considered a design-basis event for the plant. This distance is beyond the radius of peak incident pressure of 1 psi given in Regulatory Guide 1.91.

Commercial air service is provided at the Clarence E. Hancock Airport, located 49.8 km from the site. The nearest flight corridor is 22.2 km from the site. Light plane traffic is handled at Oswego County Airport, approximately 19.3 km from the site. Lakeside Airstrip, a private facility which operates as a maintenance facility with little air traffic, is located approximately 10 km from the site. Helicopter service is provided from the Hancock Airport to the NMP-2 site. The service approaches within 1000-2000 feet west and south of the Unit 2 reactor building. The aircraft crash frequency is considered to be below 10^{-7} per year, since the annual aircraft movements are below the critical number for which a probability analysis would be required per Regulatory Guide 1.70. The helicopter crash frequency has been conservatively estimated to be 10^{-6} per year using SRP 2.2.3 methodology. The licensee states that, in accordance with the SRP, "additional qualitative arguments could be made which would lower this probability to less than about 10^{-7} per year," and that this satisfies the requirements of Regulatory Guide 1.70 such that helicopter crashes need not be considered a design-basis accident.

A propane storage tank at the Fitzpatrick plant contains approximately 1,000 gallons of propane. The licensee analyzed the peak pressure from delayed ignition of a vapor cloud, and found that this event would not cause a 1-psi overpressure to reach the Unit 2 containment building. (The 1-psi criterion is from Regulatory Guide 1.91.)

Sithe Energies, USA, has recently completed construction of the Independence electrical generating station 2 miles from the NMP-2 site. This plant is a natural gas-fired generating station. A natural gas pipeline serving this facility passes within 2 miles of NMP-2. NMPC performed a calculation considering the consequences of a postulated pipe break in this natural gas line, with a ground level release at sonic velocity at the point closest to NMP-2. The maximum resulting pressure was determined to be less than 1 psi.

2.3.3.3 Significant Changes Since Issuance of the Operating License

The submittal did not identify any significant changes since issuance of the operating license.

2.3.3.4 Significant Findings and Plant-Unique Features

No significant findings were cited in the submittal.

2.3.3.5 Hazard Frequency

In summary, the licensee reviewed and identified potential sources of transportation and nearby facility accidents, and all of the identified sources screened out.

2.3.4 Other HFO Events

Beyond the events evaluated above, the licensee's IPEEE submittal provides little additional information. The licensee's approach to identifying other external events was not comprehensive, and did not rely on well-established methods [9] for performing such an analysis. Some arguments are presented in support of screening lightning, severe temperature transients, severe weather storms, external fires, extraterrestrial activity, and volcanic activity. This is a limited set of a much broader listing of external events contained in the NRC *PRA Procedures Guide* [17]. More recent assessments build on the PRA Procedures Guide

approach, which has been widely used and peer reviewed. If the licensee did, in fact, use such an approach to conduct a systematic search for possible external initiators, its basis and results are not documented in the IPEEE submittal. Thus, it is not possible at this stage in the review to conclude that there are no other external initiators of concern for NMP-2.

2.4 Generic Safety Issues (GSI-147, GSI-148, and GSI-172)

2.4.1 GSI-147, "Fire-Induced Alternate Shutdown/Control Panel Interaction"

GSI-147 addresses the scenario of fire occurring in a plant (e.g., in the control room), and conditions which could develop that may create a number of potential control system vulnerabilities. Control system interactions can impact plant risk in the following ways:

- Electrical independence of remote shutdown control systems
- Loss of control power before transfer
- Total loss of system function
- Spurious actuation of components

The licensee evaluated hot shorts leading to LOCAs and interfacing system LOCAs. All circuitry associated with remote shutdown was found to be electrically independent of the control room. The submittal has followed the guidance provided in FIVE concerning control system interactions; however, little detail has been provided.

2.4.2 GSI-148, "Smoke Control and Manual Fire Fighting Effectiveness"

GSI-148 addresses the effectiveness of manual fire-fighting in the presence of smoke. Smoke can impact plant risk in the following ways:

- By reducing manual fire-fighting effectiveness and causing misdirected suppression efforts
- By damaging or degrading electronic equipment
- By hampering the operator's ability to safely shutdown the plant
- By initiating automatic fire protection systems in areas away from the fire

Reference [18] identifies possible reduction of manual fire-fighting effectiveness and causing misdirected suppression efforts as the central issue in GSI-148. Manual fire fighting was credited only in the control room analysis. No specific information was provided concerning the potential for smoke to reduce manual fire-fighting effectiveness or misdirect suppression efforts.

2.4.3 GSI-172, "Multiple System Responses Program (MSRP)"

Reference [18] provides the description of each MSRP issue stated below, and delineates the scope of information that may be reported in an IPEEE submittal relevant to each such issue. The objective of this subsection is only to identify the location in the IPEEE submittal where information having potential relevance to GSI-172 may be found.

Common Cause Failures (CCFs) Related to Human Errors

Description of the Issue [18]: CCFs resulting from human errors include operator acts of commission or omission that could be initiating events, or could affect redundant safety-related trains needed to mitigate the events. Other human errors that could initiate CCFs include: manufacturing errors in components that affect redundant trains; and installation, maintenance or testing errors that are repeated on redundant trains. In IPEEEs, licensees were requested to address only the human errors involving operator recovery actions following the occurrence of external initiating events.

Sections 3.1.2.1 (page 3.1-6), 3.1.2.1.6 (detailed discussion of human actions screening for the SMA), 3.1.5, 3.2.5.1, and 3.2.5.2 (with discussion of human response considerations for the seismic PRA provided on page 3.2-18) of the NMP-2 IPEEE submittal provide information on the treatment of operator recovery actions for the seismic analysis. In regard to the fire analysis, the submittal provides information on operator recovery actions in Sections 1.4 (pages 1-10 and 1-11), 4.6.2.2 (pages 4.6-14 and 4.6-15), and 4.6.2.3 (pages 4.6-15 to 4.6-58). For the HFO events analysis, operator recovery actions in the event of a flood are discussed in Section 5.2.4.

Non-Safety-Related Control System/Safety-Related Protection System Dependencies

Description of the Issue [18]: Multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems, as a result of potential unrecognized dependencies between control and protection systems. The concern is that plant-specific implementation of the regulations regarding separation and independence of control and protection systems may be inadequate. The licensees' IPE process should provide a framework for systematic evaluation of interdependence between safety-related and non-safety-related systems, and should identify potential sources of vulnerabilities. The dependencies between safety-related and non-safety-related systems resulting from external events -- i.e., concerns related to spatial and functional interactions -- are addressed as part of "fire-induced alternate shutdown and control room panel interactions," GSI-147, for fire events, and "seismically induced spatial and functional interactions" for seismic events.

Information provided in the NMP-2 IPEEE submittal pertaining to seismically induced spatial and functional interactions is identified below (under the heading *Seismically Induced Spatial and Functional Interactions*), whereas information pertaining to fire-induced alternate shutdown and control panel interactions has already been identified in Section 2.4.1 of this TER.

Heat/Smoke/Water Propagation Effects from Fires

Description of the Issue [18]: Fire can damage one train of equipment in one fire zone, while a redundant train could potentially be damaged in one of following ways:

- Heat, smoke, and water may propagate (e.g., through HVAC ducts or electrical conduit) into a second fire zone, and damage a redundant train of equipment.

- A random failure, not related to the fire, could damage a redundant train.

- Multiple non-safety-related control systems could be damaged by the fire, and their failures could affect safety-related protection equipment for a redundant train in a second zone.

A fire can cause unintended operation of equipment due to hot shorts, open circuits, and shorts to ground. Consequently, components could be energized or de-energized, valves could fail open or closed, pumps could continue to run or fail to run, and electrical breakers could fail open or closed. The concern of water propagation effects resulting from fire is partially addressed in GI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment." The concern of smoke propagation effects is addressed in GSI-148. The concern of alternate shutdown/control room interactions (i.e., hot shorts and other items just mentioned) is addressed in GSI-147.

Information provided in the NMP-2 IPEEE submittal pertaining to GSI-147 and GSI-148 has already been identified in Sections 2.4.1 and 2.4.2 of this TER. Sections 4.8.4 of the submittal presents some information pertaining to this issue.

Effects of Fire Suppression System Actuation on Non-Safety-Related and Safety-Related Equipment

Description of the Issue [18]: Fire suppression system actuation events can have an adverse effect on safety-related components, either through direct contact with suppression agents or through indirect interaction with non-safety related components. This concern is addressed in GI-57.

Information pertaining to suppression-induced damage to equipment, as well as seismically induced inadvertent actuation of fire suppression systems, can be found, respectively, in Section 4.8.4, and in Sections 4.8.1.2 and 4.8.1.3, of the IPEEE submittal.

Effects of Flooding and/or Moisture Intrusion on Non-Safety-Related and Safety-Related Equipment

Description of the Issue [18]: Flooding and water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment. This type of event can result from external flooding events, tank and pipe ruptures, actuations of fire suppression systems, or backflow through parts of the plant drainage system. The IPE process addresses the concerns of moisture intrusion and internal flooding (i.e., tank and pipe ruptures or backflow through part of the plant drainage system). The guidance for addressing the concern of external flooding is provided in Chapter 5 of NUREG-1407, and the concern of actuations of fire suppression systems is provided in Chapter 4 of NUREG-1407.

The following information is provided relevant to this issue: the NMP-2 IPEEE submittal discusses external floods in Section 5.2; discussion is provided in Section 4.8.4 regarding actuations of fire suppression systems; discussion of seismically induced inadvertent actuation of fire suppression systems is provided in Sections 4.8.1.2 and 4.8.1.3; and discussion on seismically induced flooding is provided in Sections 3.1.1 and 3.1.2.1.5, in addition to Sections 4.8.1.2 and 4.8.1.3.

Seismically Induced Spatial and Functional Interactions

Description of the Issue [18]: Seismic events have the potential to cause multiple failures of safety-related systems through spatial and functional interactions. Some particular sources of concern include: ruptures

in small piping that may disable essential plant shutdown systems; direct impact of non-seismically qualified structures, systems, and components that may cause small piping failures; seismic functional interactions of control and safety-related protection systems via multiple non-safety-related control systems' failures; and indirect impacts, such as dust generation, disabling essential plant shutdown systems. As part of the IPEEE, it was specifically requested that seismically induced spatial interactions be addressed during plant walkdowns. The guidance for performing such walkdowns can be found in EPRI NP-6041.

The NMP-2 IPEEE has included a seismic walkdown which investigated the potential for adverse physical interactions. The submittal states that EPRI NP-6041 guidelines were followed in the seismic walkdowns. Relevant information can be found in Sections 3.1.1, 3.1.2.1.5, 3.1.5, 3.2.2, and 4.8.1.

Seismically Induced Fires

Description of the Issue [18]: Seismically induced fires may cause multiple failures of safety-related systems. The occurrence of a seismic event could create fires in multiple locations, simultaneously degrade fire suppression capability, and prevent mitigation of fire damage to multiple safety-related systems. Seismically induced fires is one aspect of seismic-fire interaction concerns, which is addressed as part of the Fire Risk Scoping Study (FRSS) issues. (IPEEE guidance specifically requested licensees to evaluate FRSS issues.) In IPEEEs, seismically induced fires should be addressed by means of a focused seismic-fire interactions walkdown that follows the guidance of EPRI NP-6041.

Section 4.8.1.1 of the NMP-2 IPEEE submittal provides a discussion of seismically induced fires.

Seismically Induced Fire Suppression System Actuation

Description of the Issue [18]: Seismic events can potentially cause multiple fire suppression system actuations which, in turn, may cause failures of redundant trains of safety-related systems. Analyses currently required by fire protection regulations generally only examine inadvertent actuations of fire suppression systems as single, independent events, whereas a seismic event could cause multiple actuations of fire suppression systems in various areas.

Sections 4.8.1.2 and 4.8.1.3 of the NMP-2 IPEEE submittal provide discussion of seismically induced fire suppression system actuation and degradation.

Seismically Induced Flooding

Description of the Issue [18]: Seismically induced flooding events can potentially cause multiple failures of safety-related systems. Rupture of small piping could provide flood sources that could potentially affect multiple safety-related components simultaneously. Similarly, non-seismically qualified tanks are a potential flood source of concern. IPEEE guidance specifically requested licensees to address this issue.

The NMP-2 IPEEE submittal includes information on seismically induced flooding in Sections 3.1.1 and 3.1.2.1.5, as well as in Sections 4.8.1.2 and 4.8.1.3.

[illegible]

Seismically Induced Relay Chatter

Description of the Issue [18]: Essential relays must operate during and after an earthquake, and must meet one of the following conditions:

- remain functional (i.e., without occurrence of contact chattering);
- be seismically qualified; or
- be chatter acceptable.

It is possible that contact chatter of relays not required to operate during seismic events may produce some unanalyzed faulting mode that may affect the operability of equipment required to mitigate the event. IPEEE guidance specifically requested licensees to address the issue of relay chatter.

The NMP-2 IPEEE submittal provides extensive discussion pertaining to the evaluation of seismically induced relay/contactor chatter. A detailed analysis of relay chatter is documented in Section 3.1.2.2 of the submittal. In addition, Section 3.1.5 discusses an evaluation of the effects of relay chatter on containment isolation, Section 3.2 includes consideration of relay chatter effects in the seismic PRA, and Section 4.8.1.2 discusses the effects of relay chatter on seismically induced inadvertent actuation of fire suppression systems.

Evaluation of Earthquake Magnitudes Greater than the Safe Shutdown Earthquake

Description of the Issue [18]: The concern of this issue is that adequate margin may not have been included in the design of some safety-related equipment. As part of the IPEEE, all licensees are expected to identify potential seismic vulnerabilities or assess the seismic capacities of their plants either by performing seismic PRAs or seismic margins assessments (SMAs). The licensee's evaluation for potential vulnerabilities (or unusually low plant seismic capacity) due to seismic events should address this issue.

The NMP-2 IPEEE has included both a seismic margin assessment and a seismic PRA, as documented in Section 3 of the submittal. The seismic input for the analysis is described in Section 3.1.3 of the submittal.

Effects of Hydrogen Line Ruptures

Description of the Issue [18]: Hydrogen is used in electrical generators at nuclear plants to reduce windage losses, and as a heat transfer agent. It is also used in some tanks (e.g., volume control tanks) as a cover gas. Leaks or breaks in hydrogen supply piping could result in the accumulation of a combustible mixture of air and hydrogen in vital areas, resulting in a fire and/or an explosion that could damage vital safety-related systems in the plants. It should be anticipated that the licensee will treat the hydrogen lines and tanks as potential fixed fire sources as described in EPRI's FIVE guide, assess the effects of hydrogen line and tank ruptures, and report the results in the fire portion of the IPEEE submittal.

Information on hydrogen line ruptures is discussed in Sections 4.1.3.2 (page 4.1-9) and 4.8.1.1. Some information on hydrogen storage is also provided in Section 5.1.4.

3 OVERALL EVALUATION AND CONCLUSIONS

3.1 Seismic

3.1.1 Seismic Margin Assessment

As implemented by the licensee, the seismic margin assessment (SMA) consisted of determining HCLPF values for components in a simplified success path. The HCLPF capacity determination was extended to provide seismic fragility results, since most of the work necessary to define fragilities had already been completed. A review level earthquake (RLE) of 0.5g PGA was used for screening, rather than the value of 0.3g PGA recommended in NUREG-1407 [3]. In addition, the licensee stated that a seismic PRA was performed since it was a "relatively insignificant effort with all the inputs already available." For quantification of the PRA results the licensee employed seismic hazard estimates prepared by EPRI [19] and LLNL [20]. The simplified seismic PRA employed a surrogate element which represented the screened-out structures, systems, and components.

The strengths and weaknesses of the SMA evaluation, as discovered in this review, are as follows:

Strengths

1. The licensee used an approved methodology (EPRI seismic margins method) and conservatively applied the methodology at 0.5g PGA for the review level earthquake (compared with the NRC request for an evaluation at 0.3g PGA).

2. The licensee, recognizing that the incremental work was not significant, extended the SMA work to a simplified seismic PRA in order to obtain additional insights.

3. The walkdown team's composition and member qualifications were well documented, and the walkdown process was consistent with EPRI NP-6041 (Rev. 1) guidelines, and NUREG-1407 procedures.

4. The licensee's relay chatter evaluation was expanded beyond that requested for a focused-scope 0.3g PGA plant. Instead of limiting the scope to bad-actor relays, the licensee reviewed all relays within the preferred and alternate shutdown paths.

Weaknesses

1. The licensee's use of HPCS and RCIC as alternate success paths is contrary to the EPRI NP-6041 guidelines which the licensee adopted for the SMA. The individual HPCS and RCIC failure rates, as well as the combined unreliability of these two systems for the high-pressure coolant makeup function, exceed EPRI NP-6041 guidelines.

3.1.2 Seismic Probabilistic Risk Assessment

The licensee performed a seismic PRA to put the SMA results into perspective and to provide additional public-safety and economic-risk insights (by providing, together with the IPE, a more complete risk model

and tool for decision making). Moreover, the licensee concluded that, with the availability of the NMP-2 IPE model, the SMA, the EPRI and LLNL seismic hazard results, and insights observed from other seismic PRAs, the additional effort to perform this PRA was comparatively insignificant.

The licensee concluded that, since the seismic hazard was low and the plant HCLPF capacity was high, the risk associated with seismic events was expected to be very low. Thus, the licensee argued that simplifying conservative assumptions could be made to estimate risk.

Seismic hazard results from EPRI [19] and LLNL [20] were used to quantify the frequency of core damage and radiological releases. Seismic fragilities from the SMA were used, with the exception of use of a generic loss of offsite power fragility from the Seabrook seismic PRA [21]. Non-seismic unavailabilities for systems and functions having relatively high unavailabilities were obtained from the NMP-2 IPE. Using these inputs, a simplified event-tree model was developed from the IPE, and the RISKMAN code was used to quantify the model.

Three endpoints were numerically evaluated using this model: (a) late core melt due to loss of heat removal or loss of injection, (b) early core melt due to loss of injection with the containment isolated, and (c) early core melt due to loss of injection with the containment unisolated. The results of the study were expressed using both the EPRI and LLNL seismic-hazard inputs, as shown in Table 3.1 below.

Table 3.1 Nine Mile Point Unit 2 Seismic PRA Results

Core Damage Timing and Containment Status	Mean Annual Frequency	
	EPRI	LLNL
Late (Loss of Heat Removal or Injection)	5.9×10^{-8}	2.2×10^{-7}
Early (Loss of Injection, Isolated Containment)	3.2×10^{-8}	9.9×10^{-8}
Early (Loss of Injection, Unisolated Containment)	1.6×10^{-7}	9.0×10^{-7}
TOTAL Seismic PRA CDF	2.5×10^{-7}	1.2×10^{-6}

The licensee makes note of the following insights derived from the seismic PRA of NMP-2:

- All safety-related equipment in the SMA success path was assessed to have a HCLPF capacity of at least 0.5g PGA. This plant HCLPF was modeled as a direct cause of core damage (early core damage, with the containment unisolated). The plant HCLPF was responsible for 65% of the seismic PRA CDF using the EPRI hazard results, and 75% of the CDF using the LLNL hazard results.

- Failure of the non-safety-related (non-seismically qualified) high pressure nitrogen accumulators was assessed as having a HCLPF capacity of 0.23g PGA. The seismic PRA model assumes that low-pressure injection fails in the long term if nitrogen fails (consistent with the SMA success path). This scenario dominates the frequency of the late core damage end state (see Table 3.1),

and is responsible for 15% of the seismic CDF based on EPRI hazard input and 13% of the seismic CDF based on the LLNL hazard input. Although the licensee performed the SMA at a 0.5g review level and states that the plant demonstrates a 0.5g HCLPF, this is only valid for 24 hours. The licensee could only demonstrate a 72-hour HCLPF (which is the basis for SMA studies identified in EPRI-NP-6041, Rev. 1) of 0.23g.

- Station blackout was assessed as contributing 15% of the seismic CDF for EPRI hazard input, and 8.3% of the seismic CDF for LLNL hazard input. (No recovery is modeled for station blackout.)

- Early core melt with unisolated containment was assessed as contributing 14.8% of the seismic CDF for EPRI hazard input, and 9.1% of the seismic CDF for LLNL hazard input (assuming no operator recovery).

In the present review, it was noted that the seismic hazard assessment (in both the EPRI and LLNL cases) was truncated at 1.02g. No sensitivity analysis or justification is provided for this truncation, contrary to the guidance in NUREG-1407 (which recommended use of at least 1.5g as a truncation guideline).

3.2 Fire

The licensee has expended significant effort in the preparation of the fire-analysis portion of the IPEEE. For the most part, the IPEEE submittal complies with the conditions set forth in Reference [2]. In general, the licensee has employed proper methodology and databases for conducting the fire analysis. The analysis combines methods of previous fires PRAs with the FIVE methodology.

The following items are identified as the primary strengths and weaknesses of the submittal:

Strengths

1. The submittal is very well written. The overall presentation is clear and well organized. There are sufficient tables and figures to provide the necessary information to support the analyses and the conclusions.

2. The final CDF can be traced back to the initial assumptions and frequencies. The reviewer was able to trace some of the calculations through the analysis.

3. The study has done a good job of qualitatively addressing uncertainties. The study event-tree/fault-tree methodology is sound, and the selection of initiating events appears reasonable. The study's logic in the development of initiating event frequencies, and in combining fire frequencies with random failures, is sound.

Weaknesses

1. The operator recovery probabilities for the control room fire scenarios are highly optimistic.

2. The heat release rate for an electrical cabinet fire is assumed to be 65 Btu/sec, and is not representative of cabinet fire-test data.

3. The potential adverse effects on plant safety-related equipment due to combustion products have not been adequately addressed.

4. The potential for cross-zone fire and smoke spread was not considered.

3.3 HFO Events

The licensee performed the analysis of high winds and flooding in accordance with the NUREG-1407 progressive screening approach, resulting in these events being screened out based on conformance with 1975 SRP criteria. The discussion on flooding makes it apparent that, despite SRP conformance, there are no bounding PRA arguments that can be made to dismiss external flooding as a possible contributor (due, primarily, to the existence of large uncertainties in the frequency of flooding above a critical elevation, which would cause failure of the emergency switchgear). The licensee's rationale for dismissing external flooding as a contributor is incomplete.

4 IPEEE INSIGHTS, IMPROVEMENTS, AND COMMITMENTS

4.1 Seismic

The licensee concluded that the IPEEE SMA identified no vulnerabilities or outliers. The licensee identified no substantial improvements nor commitments related to the seismic margin assessment or the seismic probabilistic risk assessment. During the walkdowns, a question was raised regarding a storage rack near a RCIC motor-operated valve. A recommendation to secure the storage rack was implemented.

4.2 Fire

The licensee has stated: "The additional contribution from this study of external events is approximately 1E-6 per year from fires. These results suggest that operation of NMP-2 poses no undue risk to the public and is within the range of CDFs for other nuclear plants. In addition to the evaluation of accident sequences that could lead to core damage, the NMP-2 IPEEE has also evaluated containment performance. The containment evaluation indicated that the NMP-2 containment does not have any unusual characteristics that result in poor containment performance."

As a result, no major safety enhancements have been identified, and consequently, no commitments are made that would require tracking by the NRC.

The entire fire IPEEE investigation, of course, has provided an excellent opportunity for the licensee's engineers to better learn about the characteristics of the plant, how the plant would behave under fire conditions, and what human actions would be necessary to protect the reactor core from any adverse effects.

4.3 HFO Events

No vulnerabilities or outliers were identified among the HFO events. No plant improvements were identified or committed to by the licensee with respect to HFO events.

5 IPEEE EVALUATION AND DATA SUMMARY SHEETS

Completed data entry sheets for the NMP-2 IPEEE are provided in Tables 5.1 to 5.7. These tables have been completed in accordance with the descriptions in Reference [7]. Table 5.1 lists the overall external events results. Table 5.2 summarizes general seismic data pertaining to the focused-scope seismic evaluation. Table 5.3 provides the BWR Seismic Success Paths table, which gives a description of the success paths developed for the focused-scope seismic evaluation. Tables 5.4 and 5.5, respectively, present BWR Accident Sequence Overview and BWR Accident Sequence Detailed tables for the seismic PRA. Tables 5.6 and 5.7, respectively, present BWR Accident Sequence Overview and BWR Accident Sequence Detailed tables for the fire PRA.

Table 5.1
External Events Results

Plant Name: Nine Mile Point Unit 2

Sheet 1 of 1

Event	Screening	CDF	Plant HCLPF(g)	Notes
External Fire (Forest Fire)	O			
External Flooding	O	"negligible"		
Extreme Winds	O	9.5×10^{-8}		
Internal Fire	S	1.4×10^{-6}		control room fires only; others screened
Nearby Facility Accidents	O			
Seismic Activity	S	2.5×10^{-7} (EPRI) 1.2×10^{-6} (LLNL 1994)	0.5	
Transportation Accidents	O	"less than about 10^{-7}"		
Hail	O			
Lightning	O			
Turbine Missiles	O			
Ice and Snow	O			

Screening: S = Plant specific analysis; O = Screened out; SO = Bounding analysis

Table 5.2
SMM Seismic Fragility

Plant Name: Nine Mile Point Unit 2

Review Level Earthquake (g): 0.5g

Spectral Shape: NUREG/CR-0098
(NUREG/CR-0098, NRC Guide 1.60, 10,000 year LLNL median UHS, Site Specific, or other)

List components and equipments which do not meet RLE (all components) or with lowest HCLPF (less than 10):

Component	HCLPF (g)	Seismic Sequence Description	Seismic Success Path Description
ADS Nitrogen Tanks	0.23		Not on Success Path

Table 5.3
BWR Seismic Success Paths

Plant Name: Nine Mile Point Unit 2

1 Sheet of 1

CHALLENGE	STRATEGY	SUCCESS PATH	REACTIVITY CONTROL					PRESSURE BOUNDARY INTEGRITY					HIGH PRESSURE INJECTION						LOW PRESSURE INJECTION						CONTAINMENT SYSTEMS											NOTES
			RPS	ARI	SLC	CRDS	RECIRC	SRVS	ADS	MSIV	ADEP	TBV	HPCI/HPCS	RCIC	RWCU	MFW	HP1	HP2	LPCI	CS	CTS	LP1	LP2	LP3	SPC	DWS	DWC	ASPC	ICS	CI1	CI2	INERT	DWIGN	WWIGN	HUM	
T-LOOP			X					X						X											X											No low pressure injection
T-LOOP			X					X					X												X											No low pressure injection

Challenge: One of the following: S1, S2, S3, A, V(-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-RECIRC, T-LNMU, T-LMFW, T-EXFW, T-SLBOC, T-SLBIC, T-SORV/IORV, T-SSI, T-(Other), OR T-(Support System). (-xx) refers to optioanl supplementary material.

Acronym of Support Systems: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC4, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, NSW, OA3, 0A4, RBCLCW, SA, STM, SW2, SW3, SW4, TBCLCW, VAC

1,2,3...How many needed to operate H = Human action required T = Must be throttled/controlled

For Core Damage Prevention Challenges, show only hardware whose failure is modeled as contributing to core damage.

Table 5.4
BWR Accident Sequence Overview Table

Plant Name: Nine Mile Point Unit 2   For Seismic PRA Only   1 Sheet of 1

#	Sequence	PDS	CDF	Init. Event	Lost Supports	Failed Functions	Attributes
1	Unisolated Containment-Loss of Injection		1.6×10^{-7} (EPRI) 9.0×10^{-7} (LLNL)	T-LOOP	LPI, VENT, RCS-DEP	HPI	
2	Late-Loss of Heat Removal or Injection		5.9×10^{-8} (EPRI) 2.2×10^{-7} (LLNL)	T-LOOP	LPI, VENT, RCS-DEP	CPSR	
3	Isolated Containment-Loss of Injection		3.2×10^{-8} (EPRI) 9.9×10^{-8} (LLNL)	T-LOOP	LPI, VENT, RCS-DEP	HPI	

Init. Event (Initiator): One of the following: S1, S2, S3, A, V (-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-RECIRC, T-LNMU, T-LMFW, T-EXFW, T-SLBOC, T-SLBIC, T-SORV/IORV, T-SSI, T-(Other), or T-(Acronym)
(-xx) refers to optional supplementary material.

Lost Supports: At most two of the following: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC4, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, NSW, OA3, OA4, RBCLCW, SA, STM, SW2, SW3, SW4, TBCLCW, VAC (Field may be blank).

Failed Functions: At most three of the following: RCS-BOR, RCS-INT, RCS-DEP, HPI, HPR, LPI, LPR, CPSI, CPSR, CIF, VENT (If a 4th and/or 5th are necessary, use the "Notes" field)

Attributes: At most three of the following: ATWS, BYPASS, TH., SBO, OR HUM (Field may be blank)

Table 5.5
BWR Accident Sequence Detailed Table

Plant Name: Nine Mile Point Unit 2

For Seismic PRA only

#	SEQUENCE	REACTIVITY CONTROL					PRESSURE BOUNDARY INTEGRITY					HIGH PRESSURE INJECTION						LOW PRESSURE INJECTION						CONTAINMENT SYSTEMS											NOTES
		RPS	ARI	SLC	CRDS	RECIRC	SRVS	ADS	MSIV	ADEP	TBV	HPCI/HPCS	RCIC	RWCU	MFW	HP1	HP2	LPCI	CS	CTS	LP1	LP2	LP3	SPC	DWS	DWC	ASPC	ICS	CI1	CI2	INERT	DW IGN	WW IGN	HUM	
1	Unisolated Containment-Loss of Injection	X					X					X	X		X			X	X	X	X	X	X						X	X					Unisolated Containment-Early Melt
2	Late-Loss of Heat Removal or Injection	X					X								X			X	X	X	X	X	X	X	X										Late Melt
3	Isolated Containment-Loss of Injection	X					X					X	X		X			X	X	X	X	X	X												Isolated Containment-Early Melt

Table 5.6
BWR Accident Sequence Overview Table

Plant Name: Nine Mile Point Unit 2 | For Fire PRA Only | 1 Sheet of 1

#	Sequence	PDS	CDF	Init. Event	Lost Supports	Failed Functions	Attributes
1	Control Room Fire-Abandonment		$6.2x10^{-7}$	T-LMFW		RCIC, HPCS, ADEP	HUM
2	Panel 852 Fire-Station Blackout		$3.5x10^{-7}$	T-LOOP		RCIC, EAC	
3	Panel 852 Fire-Loss of Injection		$2.2x10^{-7}$	T-LOOP		RCIC HPCS, ADEP	HUM
4	Panel 601 Fire-Loss of Injection		$1.1x10^{-7}$	T-LMFW		RCIC, HPCS, ADEP	HUM

Init. Event (Initiator): One of the following: S1, S2, S3, A, V (-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-RECIRC, T-LNMU, T-LMFW, T-EXFW, T-SLBOC, T-SLBIC, T-SORV/IORV, T-SSI, T-(Other), or T-(Acronym)
(-xx) refers to optional supplementary material.

Lost Supports: At most two of the following: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC4, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, NSW, OA3, OA4, RBCLCW, SA, STM, SW2, SW3, SW4, TBCLCW, VAC (Field may be blank).

Failed Functions: At most three of the following: RCS-BOR, RCS-INT, RCS-DEP, HPI, HPR, LPI, LPR, CPSI, CPSR, CHF, VENT (If a 4th and/or 5th are necessary, use the "Notes" field)

Attributes: At most three of the following: ATWS, BYPASS, TIL, SBO, OR HUM (Field may be blank)

[illegible]

Table 5.7
BWR Accident Sequence Detailed Table

Plant Name: Nine Mile Point Unit 2

For Fire PRA only

		REACTIVITY CONTROL					PRESSURE BOUNDARY INTEGRITY					HIGH PRESSURE INJECTION						LOW PRESSURE INJECTION						CONTAINMENT SYSTEMS											
#	SEQUENCE	RPS	ARI	SLC	CRDS	RECIRC	SRVS	ADS	MSIV	ADEP	TBV	HPCI/HPCS	RCIC	RWCU	MFW	HP1	HP2	LPCI	CS	CTS	LP1	LP2	LP3	SPC	DWS	DWC	ASPC	ICS	CI1	CI2	INERT	DW IGN	WW IGN	HUM	NOTES
1	Control Room Fire-Abandonment							X		X		X	X		X																			X	Control Room Fire-Remote Shutdown
2	Panel 852 Fire-Station Blackout							X		X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X								Station Blackout
3	Panel 852 Fire-Loss of Injection									X		X	X																					X	
4	Panel 601 Fire-Loss of Injection									X		X	X																					X	

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21. "Individual Plant Examination of External Events, Report for Seabrook Station, Response to Generic Letter 88-20, Supplement 4," North Atlantic Energy Services Corporation, September 1992.