ERI/NRC 95-514

TECHNICAL EVALUATION REPORT ON THE "SUBMITTAL-ONLY" REVIEW OF THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS AT KEWAUNEE NUCLEAR POWER PLANT

FINAL REPORT

Completed: November 1995
Revised: March 1996 and January 1998
Final: January 1999

Energy Research, Inc.
P.O. Box 2034
Rockville, Maryland 20847-2034

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

Attachment

ERI/NRC 95-514

TECHNICAL EVALUATION REPORT ON THE "SUBMITTAL-ONLY" REVIEW OF THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS AT KEWAUNEE NUCLEAR POWER PLANT

FINAL REPORT

Completed: November 1995
Revised: March 1996 and January 1998
Final: January 1999

M. Khatib-Rahbar
Principal Investigator

Authors:

R.T. Sewell[1], A.S. Kuritzky, H. Esmaili,
M. Kazarians[2], A. Mosleh[3], M.V. Frank[4], and R.J. Budnitz[5]
Energy Research, Inc.
P.O. Box 2034
Rockville, Maryland 20847

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

1 Present address: EQE International, 2942 Evergreen Parkway, Suite 302, Evergreen, CO 80439

2 Kazarians and Associates, 425 East Colorado Street, Suite 545, Glendale, CA 91205

3 University of Maryland, Dept. of Materials and Nuclear Engineering, College Park, MD 20742

4 Safety Factor Associates, Inc., 1410 Vanessa Circle, Suite 16, Encinitas, CA 92024

5 Future Resources Associates, Inc., 2039 Shattuck Ave., Suite 402, Berkeley, CA 94704

TABLE OF CONTENTS

LIST OF TABLES

EXECUTIVE SUMMARY

This technical evaluation report (TER) documents a "submittal-only" review of the Individual Plant Examination of External Events (IPEEE) conducted for the Kewaunee Nuclear Power Plant. 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 Kewaunee 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.

Wisconsin Public Service Corporation (WPSC) is the licensee of Kewaunee Nuclear Power Plant (also denoted as Kewaunee in this TER). The Kewaunee IPEEE was performed by licensee and contractor personnel. The IPEEE submittal considers seismic, fire, and HFO (high wind, flood, and other external event) initiators for the external events analysis. Table E.1 provides a summary description of the IPEEE process.

Licensee's IPEEE Process

Seismic

Kewaunee Nuclear Power Plant is assigned to the 0.3g focused-scope seismic review category in NUREG-1407. WPSC elected to perform a new Level-1 seismic probabilistic risk assessment (SPRA), with a qualitative and quantitative (Level-2) seismic containment analysis, for the Kewaunee IPEEE. The SPRA approach employed by WPSC is actually a composite of seismic PRA and seismic margin assessment (SMA) methods. The overall SPRA approach that was implemented generally follows the guidance described in NUREG/CR-4840, and plant seismic walkdowns were conducted using the procedures described in EPRI NP-6041 and the Generic Implementation Procedure (GIP). The SPRA makes use of a "surrogate element" to characterize the seismic capacity (fragility) of components which are screened out (based on seismic margin screening tables) at a PGA HCLPF level of roughly 0.3g. (The screening approach has followed the latest revision of EPRI NP-6041, which employs spectral-acceleration-based screening levels.) As discussed later, the use of the surrogate element for Kewaunee presents an obstacle with respect to obtaining full SPRA insights from the seismic evaluation.

Table E.1 Summary Description of the Kewaunee Nuclear Power Plant IPEEE

External Initiator	Description of Approach/Findings
Seismic	A seismic PRA was conducted for Kewaunee Nuclear Power Plant. An estimated seismic core damage frequency of 1.10×10^{-5} per reactor-year (for Electric Power Research Institute [EPRI] hazard results), and a plant-level high confidence of low probability of failure (HCLPF) capacity of 0.23g (including effects of non-seismic failures and human actions), were obtained in the study. No vulnerabilities were reported; however, a number of outliers and housekeeping concerns were noted, and have been resolved or are planned to be resolved.
Internal Fires	A combination of FIVE and PRA methodologies was implemented for Kewaunee fire IPEEE. The estimated core damage frequency from internal fire is 1.81×10^{-4} per reactor-year. No vulnerabilities have been identified and no plant improvements have been considered.
HFO Events	All HFO initiators were screened out in the IPEEE. No vulnerabilities nor plant improvements were reported as a result of the HFO evaluation.

A significant element of the seismic evaluation effort was the extensive coordination that has taken place between the USI A-46 and IPEEE programs, particularly in the walkdowns. Seismic Evaluation Work Sheets (SEWSs) were completed as part of equipment reviews.

The plant logic analysis was performed by modifying individual plant examination (IPE) event trees and fault trees. For many systems, seismic failures were addressed simply by modeling the surrogate element in series with IPE fault tree logic. Thus, the surrogate element was used to model the potential for multiple component failures that may lead to failure of the given system.

Seismic structural responses and component demands were determined using existing dynamic structural response models developed for design purposes. These models include three-dimensional lumped mass models with elastic half-space springs used to represent soil behavior. The 10,000-yr median 1989 Lawrence Livermore National Laboratory (LLNL) uniform hazard spectrum for Kewaunee was used to characterize the seismic input.

In general, the study has addressed all major elements of concern for seismic PRA evaluation of a focused-scope plant, as identified by NUREG-1407. In addition to those items just described, the study has included consideration of seismic containment performance, relay chatter evaluation, soil failures, seismic-fire interactions, and applicable Generic Issues (GIs) and Unresolved Safety Issues (USIs).

Fire

The fire analysis of the IPEEE was based on a combination of PRA and FIVE methodologies. The overall methodology, similar to other fire analysis techniques, has a graduated focus on the most important fire zones using qualitative and quantitative screening criteria. The fire zones or compartments were subjected to at least two screening stages. In the first stage, a zone was screened out if it does not contain any safety-related equipment. In the second stage, a core damage frequency (CDF) of 10^{-6} per year was used as the screening criterion.

The plant information gathered for Appendix R compliance, and other information pertinent to fire events, have been used extensively. The internal events model of the IPE has been used to establish the possibility of experiencing core damage from a fire event. The conditional core damage probability was based on the equipment and systems unaffected by the fire. The unconditional core damage frequency was obtained by multiplying the frequency of a fire in a fire zone with the conditional core damage probability for that fire zone.

For fire occurrence frequencies, for specific fire zones, the data base provided in the FIVE document has been employed. The fire frequencies were specialized for specific fire compartments, using weighting factors based on the combustible loading, type and number of components in a compartment.

For fire propagation, the COMPBRN IIIe computer program has been used. Human actions based on special fire-related procedures, and those considered in the IPE plant model, have been included in the fire impact assessment.

In addition to the fire CDF, the possibility of containment failure has been considered. Isolation failure was considered as the most significant containment failure mode, and it was found to occur for 31% of the total fire-induced CDF.

HFO Events

For HFO events, the submittal has generally followed the guidance and basic procedures of NUREG-1407 for analyzing and reporting potential accident scenarios. It used a comprehensive list of potential external hazards to identify areas where more detailed analysis were judged to be needed. These areas include High Winds and Tornadoes, External Flooding, Transportation and Nearby Facilities Accidents, and Hazardous Materials. These external events have been analyzed by a screening approach. According to the screening analysis, the contribution of HFO events to the total CDF is less than 10^{-6} per reactor-year (about 5% of the total CDF). No vulnerabilities have been identified that would require detailed quantification of any accident sequence.

Key IPEEE Findings

Seismic

In the seismic IPEEE, the overall plant HCLPF capacity has been reported to be equal to 0.23g, accounting for non-seismic failures and human errors, and 0.26g when the non-seismic failures and human errors are ignored. (These HCLPF assessments were reported with respect to the LLNL median 10,000-yr UHS shape; this spectral shape is significantly different from the NUREG/CR-0098 median, 5%-damped spectral shape which is recommended in NUREG-1407 as the basis for reporting HCLPF capacities in a seismic margin assessment.) CDF values of 1.10×10^{-5}/ry and 1.32×10^{-5}/ry have been reported, respectively, for the 1989 EPRI hazard results and the 1993 LLNL hazard results. Calculations of seismic capacities for outliers have revealed one component HCLPF estimate to be as low as 0.29g, i.e., for the residual heat removal (RHR) heat exchanger.

Twelve (12) bad-actor relays were encountered in the USI A-46 evaluation; no additional bad-actor relays were found in IPEEE-only equipment.

Overall, the submittal concludes that there are no seismic vulnerabilities at Kewaunee. However, the seismic walkdown of Kewaunee identified a number of issues which required implementation of resolution approaches. A few equipment modifications have been proposed/implemented, and a procedure to improve seismic housekeeping/maintenance has been considered.

According to the submittal, the dominant basic events/component failures that contribute to seismic risk are: loss of offsite power, failure of the surrogate element, and operator error. In addition, the licensee has arrived at the following conclusions:

a. There does not exist a single failure mode that dominates the seismic core damage frequency.

b. Failure of the surrogate element is, for many systems, the important mode of failure. Thus, the fact that the surrogate element does not specifically model the failure of a particular component, further reinforces the conclusion that there are no specific component failures that dominate the seismic CDF.

c. Operator actions are not a major contributor to the seismic CDF or plant capacity.

d. Loss of offsite power is an important contributor to the seismic risk.

e. As a group, random failures and operator actions are an important part of the seismic CDF.

f. For seismic containment performance, the results of the SPRA evaluations indicate that the containment, as well as the systems designed to ensure containment integrity, are seismically sound, and no vulnerabilities could be identified.

For reasons discussed in this TER, many of these conclusions of the seismic IPEEE are not considered to be highly meaningful.

Fire

For fire events, the CDF was estimated at 1.81×10^{-4}/ry. This value is within the range of frequencies typically reported in fire PRAs and IPEEEs. There are six scenarios with CDFs greater than 10^{-5}/ry, and one scenario with CDF greater than 10^{-6}/ry. According to the submittal, most significant core damage sequences include auxiliary feedwater system and bleed and feed failures. The scenarios that have a CDF greater than 10^{-5}/ry consist of fires in the two auxiliary feedwater pump rooms, the cable spreading room, one of the diesel rooms and the control room.

The licensee does not suggest specific fire-related improvements based on the IPEEE final results. The licensee claims that the final results are conservative, but does not attempt to conduct sensitivity analyses to assess the levels of conservatism. From the descriptions provided in the submittal, it can be concluded that some scenarios can be deemed as conservative, and others are within the range of CDF values expected for a plant of similar design features.

Containment failure has been addressed and it is concluded that fires can only affect containment isolation capability.

The entire exercise of performing a fire evaluation has, of course, provided an excellent opportunity for licensee personnel to improve their knowledge of the characteristics of the plant, and how the plant would behave under fire conditions.

HFO Events

The HFO-induced CDF was estimated at a screening level of 10^{-6}/ry, which is 0.5% of the total plant CDF from both internal and external events. The HFO events which have been explicitly examined include "High Winds and Tornadoes", "External Flooding", "Transportation and Nearby Facility Accidents", and "Hazardous Materials". Key findings in these areas are summarized as follows:

1. High Winds and Tornadoes

   Kewaunee has facilities that were designed and built prior to the NRC's current criteria. Thus, the NUREG/CR-5042 approach has been used for a systematic examination of the plant. The frequency of wind load exceedance was determined to be insignificant (i.e., less than 10^{-6}/yr) and no further analysis was performed. No discussion of the potential hazard posed by tornado-generated missiles is provided in the submittal. However, the tornado missile analysis documented in the plant USAR concludes that missile impact load is unlikely to cause damage to Class I structures according to the applied design criteria. No discussion is provided as to whether or not there are non-Class I structures of importance to the IPE conclusions which may not be protected against tornado missiles.

2. External Flooding

   NUREG-0965 was used to screen out the credibility of onsite or offsite dams as potential flooding sources. Local topography, as presented in the plant's USAR, was used to argue against flooding from the landward side of the site. Thus, Lake Michigan and intense precipitation were considered as the only credible sources of external flooding. It was concluded that no flooding of the lake from a combination of rain collection and runoff will ever endanger Kewaunee.

3. Transportation and Nearby Facility Accidents

   The risk from nearby facilities was screened out on the basis that no large industrial plants are located nearby. Ground transportation accidents via road and rail were identified as the only credible source of damage from offsite hazardous materials accidents. This hazard was screened out based on lack of significant quantities of chemicals needed to cause damage. Air transportation accidents were screened out based on low hazard frequency, which was determined to be less than 10^{-7}/yr.

4. Hazardous Materials

   This analysis was basically a verification of the 1989 Updated Control Room Habitability Report, which was performed in response to NUREG/CR-0737. The submittal indicates that the analysis was further expanded to consider the effects of a release of hazardous materials on safety-related equipment or the local operation of the plant during emergencies. The submittal concludes that no vulnerabilities could be identified that would require detailed quantification of any accident sequence.

Generic Issues and Unresolved Safety Issues

Seismic

In the seismic IPEEE submittal, a detailed discussion, based on PRA methods and findings, is provided pertaining to USI A-45 resolution for external events. No plant vulnerabilities were identified as a result of the USI A-45 evaluation.

GI-131 is not, strictly speaking, applicable to Kewaunee, because the flux mapping cart is not movable. However, the lateral resistance of the mapping system was evaluated to be seismically adequate. In addition, an administrative control was implemented to insure proper restraint of a chain hoist, in order to eliminate a potential interaction hazard with the ten-path assembly of the flux mapping system.

A significant effort in coordination of USI A-46 and the seismic IPEEE has taken place for the seismic evaluation of Kewaunee. USI A-46 is resolved separately from the seismic IPEEE. The submittal notes that resolution of USI A-17 and USI A-40 will be addressed in the USI A-46 submittal.

In accordance with NUREG-1407, the Charleston Earthquake Issue is considered to be resolved with the submittal of findings from a valid seismic IPEEE.

Some information is also supplied in the IPEEE submittal which pertains to generic safety issues GSI-156 and GSI -172.

Fire

As part of the fire IPEEE submittal, the generic issues raised in the Sandia Fire Risk Scoping Study and Unresolved Safety Issue A-45 have been addressed. No vulnerabilities have been discovered as a result of this effort. Seismically-induced fires, degradation of fire suppression systems, and the possibility of inadvertent actuation of fire suppression systems as a result of an earthquake, have been addressed. The adequacy of fire barriers has been verified using fire loadings in the compartments. The plant maintains a fire brigade that conducts drills and times its response for different parts of the plant.

Some information is also supplied in the IPEEE submittal which pertains to generic safety issues GSI-147, GSI-148 and GSI-172.

HFO Events

The licensee provided information discussing the effects of rain water build-up on plant building roofing, as a result of the probable maximum precipitation (GI-103), and considers this issue resolved. Some information is also supplied in the IPEEE submittal which pertains to generic safety issues GSI-156 and GSI -172.

Vulnerabilities and Plant Improvements

Seismic

The submittal states that there are no seismic vulnerabilities at Kewaunee Nuclear Power Plant.

No major plant changes were deemed necessary by the licensee based on the results of the Kewaunee IPEEE. The seismic IPEEE did, however, identify several open issues requiring resolution. The open issues are identified in Table 3-4 of the Kewaunee IPEEE submittal (repeated in this TER as Table 4.1), together with their disposition status. Sixteen different outliers/issues are noted in the table. Some equipment enhancements, one procedural implementation, an administrative control, and several housekeeping improvements resulted from the study. The equipment enhancements included: installing missing fasteners on DG excitation and control cabinets, upgrading anchorage of station service transformers, bolting together relay racks, and implementing design changes for equipment anchorages and mercoid switches. The various plant enhancements have either been implemented or scheduled for implementation.

In one case, a HCLPF capacity was obtained which did not meet the 0.3g review level earthquake (RLE); but this item (RHR heat exchangers) was judged not to warrant a plant modification.

Fire

The licensee has concluded that there are no vulnerabilities to fires at the plant, and therefore, has not proposed any modifications affecting the fire risk.

HFO Events

The licensee has indicated that, during a safety system functional inspection of the emergency diesel generators, the design of the vents on the underground diesel oil storage were identified as an open item, and have been scheduled for resolution during 1996.

Observations

Seismic

The Kewaunee seismic IPEEE addresses the major elements specified in NUREG-1407 as recommended items that should be considered for seismic PRA evaluation of a focused-scope plant. The submittal itself gives a clear description of the seismic evaluation, and the documentation is considered to be well-written. The study provides useful information concerning dominant sequences, systems, components, and ground motions. Even though they derive principally from USI A-46 evaluation and from IPE findings, the identification and implementation of plant safety enhancements, as a result of the plant walkdowns and the IPE probabilistic safety analysis (PSA), has produced some meaningful insights in response to the objectives of GL 88-20, for a focused-scope plant. Fragility and HCLPF calculations have provided valuable information on the capability of plant components.

Even though the Kewaunee seismic IPEEE is judged to be essentially complete with respect to the guidelines and objectives of NUREG-1407 and of GL 88-20, there are some problems pertaining to implications of the licensee's seismic PRA evaluation of Kewaunee. The most significant observations/conclusions that pertain to limitations of the seismic IPEEE insights, are noted as follows:

1. The manner of usage of the surrogate element for the Kewaunee IPEEE (i.e., where there are few screened-in components having seismic capacities less than the surrogate element capacity) does not produce valid PRA insights/findings. As a consequence, a meaningful set of dominant contributors has not been found.

2. Component and plant-level HCLPF capacities are reported with respect to a UHS shape, as opposed to a NUREG/CR-0098 spectral shape (the spectral shape recommended in NUREG-1407 for reporting HCLPF capacities). The current plant HCLPF spectrum (with 0.23g or 0.26g PGA), therefore, does not exceed even the plant design spectrum over some important frequency ranges.

3. Fragilities characterizing human error rates are not realistic, and have not been based on a fundamental consideration of where and when the required human actions should be performed.

4. The study has not proposed improvements to procedures which reduce the potential for the following operator errors:

    a. Operator failure to shift auxiliary feedwater (AFW) pumps from the condensate storage tank (CST) to service water.
    b. Operator failure to open manual valve ICS-7A or ICS-7B after testing.
    c. Operator failure to initiate Internal Containment Spray (ICS) recirculation.

5. Safety enhancements to the RHR heat exchangers, and an evaluation of resulting impacts on seismic CDF, have not been considered.

Fire

With respect to the fire analysis, the licensee has certainly realized an important experience from the exercise of inspecting every part of the plant for potential fire vulnerabilities. The licensee's engineers, it can be safely claimed, have gained an excellent understanding of how the plant would behave under different fire conditions, and when human actions will be necessary to protect the plant from adverse consequences.

Overall, the licensee has employed a proper methodology and data, and the overall results are deemed to be reasonable. A thorough effort for the analysis of different issues and phenomena has been expended.

HFO Events

The HFO events portion of the submittal used a comprehensive list of potential external hazards to identify areas where more detailed analysis were judged to be needed. A mix of qualitative and quantitative arguments was used to screen out all potential accident sequences caused by HFO events. In general, the analyses are adequately supported and have followed accepted practice and the NUREG-1407 guidelines. Some specific weaknesses, however, have been identified by this review, particularly in tornado-related scenarios. These are summarized as follows:

1. The frequency of a tornado striking in the vicinity of the plant was estimated to be 4.86×10^{-4}/yr, which is above the screening level. However, the risk induced from tornadoes was screened out on the basis that the frequency of occurrence of tornadoes with wind speeds greater than the plant's design basis wind speed is negligible. No discussion of the potential hazard posed by tornado generated missiles is provided in the submittal. However, the tornado missile analysis documented in the plant USAR concludes that missile impact load is unlikely to cause damage to Class I structures according to the applied design criteria. No discussion is provided as to whether or not there are non-Class I structures of importance to the IPEEE conclusions which may not be protected against tornado missiles. The licensee has also indicated that during a safety system functional inspection of the emergency diesel generators, the design of the vents on the underground diesel oil storage had been identified as an open item, and had been scheduled for resolution during 1996.

2. The submittal states that Kewaunee has facilities that were designed and built prior to the current NRC criteria. However, such facilities were not specifically identified.

PREFACE

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

Seismic

R. Sewell, Primary Reviewer
R. Budnitz, Secondary Reviewer

Fire

M. Kazarians, Primary Reviewer
M. Frank, Secondary Reviewer

High Winds, Floods and Other External Events

A. Mosleh

Review Oversight, Coordination and Integration

M. Khatib-Rahbar, Principal Investigator
A. Kuritzky, IPEEE Review Coordination and Integration
H. Esmaili, Review and Integration
R. Sewell, Report Integration

Dr. John Lambright, of Lambright Technical Associates, contributed to the preparation of Section 2.4 following the completion of the draft version of this TER.

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

AFW	Auxiliary Feedwater
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin
CFR	Code of Federal Regnlations
CSG	Containment Safeguard
CST	Condensate Storage Tank
DBE	Design Basis Earthquake
DG	Diesel Generator
DHR	Decay Heat Removal
DSP	Dedicated Shutdown Panel
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
FSAR	Final Safety Analysis Report
GI	Generic Issue
GIP	Generic Implementation Procedure (SQUG)
GL	Generic Letter
HCLPF	High Confidence of Low Probability of Failure (Capacity)
HFO	High Winds, Floods and Other External Initiators
HVAC	Heating, Ventilation and Air Conditioning
ICS	Internal Containment Spray
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IRS	In-Structure Response Spectrum
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
MCC	Motor Control Center
MFW	Main Feedwater
MSL	Mean Sea Level
NRC	United States Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
OL	Operating License
PGA	Peak Ground Acceleration
PMP	Probable Maximum Precipitation
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RLE	Review Level Earthquake
SEWS	Seismic Evaluation Work Sheet
SI	Safety Injection

SMA	Seismic Margin Assessment
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SRT	Seismic Review Team
TER	Technical Evaluation Report
UBC	Uniform Building Code
UHS	Uniform Hazard Spectrum/Ultimate Heat Sink
USAR	Updated Safety Analysis Report
USI	Unresolved Safety Issue
USNRC	United States Nuclear Regulatory Commission
WPSC	Wisconsin Public Service Corporation

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 Kewaunee Nuclear Power Plant [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, Wisconsin Public Service Corporation (WPSC), 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 Kewaunee IPEEE submittal. Sections 2.1 to 2.3 of this report present ERI's detailed findings related to the seismic, fire, and HFO events reviews, respectively. Sections 3.1 to 3.3 summarize ERI's overall 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 the references cited in the TER.

1.1 Plant Characterization

Kewaunee Nuclear Power Plant is a single-unit, 2-loop Westinghouse Pressurized Water Reactor (PWR), with a large, dry containment of Westinghouse design. The plant is located in Kewaunee County, Wisconsin, along Lake Michigan's western shoreline. The plant commenced commercial operation on June 16, 1974. The power rating of Kewaunee is 1,650 MWt, with a net electrical output of 535 MWe. The containment at Kewaunee consists of a primary free-standing steel containment vessel, surrounded by a reinforced-concrete shield building, with an annular space between the two structures.

The Safe Shutdown Earthquake (SSE) peak ground acceleration (PGA) for Kewaunee is 0.12g; the plant Operating Basis Earthquake (OBE) has a PGA value of 0.06g. The design accelerations for vertical motions are taken to be two-thirds of the corresponding values for horizontal motions. The design spectral shape is defined by a Housner spectrum. All Class I structures/rooms have been designed for seismic loads obtained from these design motions. The turbine building and spent fuel handling area have been designed to seismic requirements of the UBC (Uniform Building Code), 1967 edition for seismic Zone 1. The auxiliary building and containment structures are founded on a common, rigid foundation mat; the turbine building and battery room are constructed on a separate rigid mat foundation. These foundations rest on clay-sand soil deposits with an approximate depth to bedrock of 76 ft.

The plant is equipped with auxiliary feedwater and charging pumps that are not dependent on external cooling. Routing of cables important to safety can be found in such areas as the auxiliary feedwater pump rooms, technical support center and diesel generator rooms. The plant is in compliance with Appendix R requirements, and all the related modifications have been completed.

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

1.2.1 Seismic

The Kewaunee seismic IPEEE was performed using seismic probabilistic risk assessment (SPRA) methodology, and has included a qualitative and quantitative (Level-2) containment performance analysis. The SPRA approach is actually a combination of SPRA and Seismic Margin Assessment (SMA) procedures. The overall Kewaunee SPRA approach follows the guidance described in NUREG/CR-4840 [4], and plant seismic walkdowns were conducted using the procedures described in Electric Power Research Institute [EPRI] NP-6041 [5] and the Generic Implementation Procedure (GIP) [6]. Plant logic models used in the SPRA were taken from the internal events individual plant examination (IPE) [7], and these were modified as necessary for seismic events. Both EPRI hazard curves and 1993 Lawrence Livermore National Laboratory (LLNL) hazard curves were used in the SPRA quantification. Peak ground acceleration (PGA) was used as the ground motion parameter. About 572 SPRA components were identified and addressed for screening evaluation. Outliers were identified, and their associated fragilities and/or High Confidence of Low Probability of Failure (HCLPF) capacities have been assessed. An interesting aspect of the SPRA is its use of a "surrogate element" to characterize the seismic capacity (fragility) of components which were screened out (based on seismic margin screening tables) at a PGA HCLPF level of roughly 0.3g. (EPRI TR-103959 [8] provides a brief description of the basis for use of the surrogate element in a seismic PRA.) The implications of the use of the surrogate element, particularly with respect to both dominant risk contributors and potential vulnerabilities, are discussed later in this review. Kewaunee is a USI A-46 plant; the USI A-46 evaluation effort was coordinated, to a significant extent, with the seismic IPEEE effort. The plant consists of a single reactor unit; hence, the IPEEE did not have to address evaluation issues pertaining to sites having multiple reactor units.

The specific elements of the Kewaunee seismic IPEEE, as described in the submittal report, include:

- Review of Plant Information
- Seismic Walkdowns
- Analysis of Plant Systems and Structural Responses
- Evaluation of Component Fragilities and Failure Modes
- Soil Liquefaction Analysis
- Relay Chatter Evaluation
- Analysis of Plant Logic and Accident Sequences
- Consideration of Non-Seismic Failures and Human Actions
- Risk Quantification and Sensitivity Analysis
- Analysis of Containment Performance
- Treatment of GI-131
- Evaluation of USI A-45

- Consideration of Seismic-Induced Fires
- Resolution of Outliers
- Peer Review
- Documentation

A number of strengths and weaknesses in the submittal's treatment of these items have been encountered in the present technical evaluation review. Related detailed observations and review findings are discussed in Section 2.1 of this TER.

The Kewaunee IPEEE submittal concludes that the plant has a HCLPF capacity of 0.23g and that the containment has a HCLPF capacity (in preventing large early failures) of 0.3g. Residual heat removal (RHR) heat exchangers were found to have a HCLPF capacity of 0.29g; they are the only components having a HCLPF capacity less than the 0.30g review level earthquake (RLE). The mean seismic core damage frequency for Kewaunee has been assessed at 1.10×10^{-5} per reactor-year (ry) for EPRI hazard input, and 1.15×10^{-5}/ry for 1993 LLNL hazard input. The mean frequency of containment failure was estimated to be 6.24×10^{-6}/ry (EPRI hazard). Outliers were identified during USI A-46/IPEEE walkdowns, and these are being addressed by meaningful safety enhancements.

Notwithstanding the weaknesses encountered in this review of the Kewaunee seismic IPEEE, it is clear that the licensee has acquired valuable information concerning the seismic capability of Kewaunee as a result of the IPEEE/USI A-46 program efforts.

1.2.2 Fire

The licensee has conducted an extensive and detailed analysis of fire events at this plant. The licensee has used state-of-the-art methodology and plant data from the Appendix R effort to conduct the analysis. Overall, the licensee has concluded that there are no significant fire vulnerabilities at Kewaunee Nuclear Power Plant. The licensee has analyzed all the fire areas of the plant using a reasonable screening methodology and PRA-based fire propagation analysis and core damage frequency evaluation model. The licensee has concluded that propagation of fires across fire zones is very unlikely and active fire dampers will function as designed. Certainly, notwithstanding the overall conclusion, the licensee has gained important experience from the exercise of inspecting every part of the plant for potential fire vulnerability. The licensee's engineers, it can be safely claimed, have gained an excellent understanding of how the plant would behave under different fire conditions, and when human actions will be necessary to protect the plant from adverse consequences.

1.2.3 HFO Events

The submittal uses a comprehensive list of potential external hazards to identify areas where more detailed analyses are judged to be needed. These areas include High Winds and Tornadoes, External Flooding, Transportation and Nearby Facilities Accidents and Hazardous Materials. These external events have been analyzed by a screening approach according to which their contribution to the total CDF is estimated to be less than 10^{-6} per year (about 5% of the total CDF). No vulnerabilities were identified that required detailed quantification of any accident sequence.

1.3 Overview of Review Process and Activities

In its qualitative review of the Kewaunee 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 includes 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 have indeed been implemented at Kewaunee

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* [9], for review of a seismic PRA, and the guidance provided in the NRC report, *IPEEE Step 1 Review Guidance Document* [10]. In addition, on the basis of the Kewaunee IPEEE submittal, ERI completed data entry tables developed in the Lawrence Livermore National Laboratory (LLNL) document entitled *"IPEEE Database Data Entry Sheet Package"* [11].

In its review of the Kewaunee seismic IPEEE, ERI examined Sections 1, 2, 3, 4.8.5, 5, 6, 7, and 8 of the IPEEE submittal [1]. The checklist of items identified in Reference [10] 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?

- Were proper methodology and data applied in the evaluation of seismic hazard, have the seismic hazard results been characterized in an appropriate way, and do the results appear reasonable, including the uncertainties in seismic hazard?

- Was the plant logic analysis performed in a manner consistent with state-of-the-art practices? Were random and human failures properly included in such analysis?

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

- Were fragility calculations performed for a meaningful set of components, and are the fragility 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 CDF?

- Are there any under-conservatisms or significant over-conservatisms in the analysis that would act to obscure dominant risk contributors and/or produce an invalid numerical estimate of CDF?

- 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 consideration of Sections 1, 2, 4, 6, 7, and 8 of Reference [1], as well as on evaluation of licensee responses (References [12], [13], and [14]) to questions presented by the NRC. The guidance provided in References [9,10] was used to formulate the review process and the organization of this document. The data entry sheets used in Section 5 have been completed in accordance with Reference [11].

The process implemented for ERI's review of the fire IPEEE included an examination of the licensee's methodology, relevant 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 and to the procedure used for estimating the frequency of occurrence of a fire scenario, in order 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 probabilistic risk assessments (PRAs). For a few fire zones/areas that were deemed important, ERI also attempted to verify the logical development of the screening justifications/arguments (especially in the case of fire-zone screening) and the computations for fire occurrence frequencies 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* [10]. This process involved examinations of the methodology, the data used, and the results and conclusions derived in the submittal. Sections 1, 2, 5, 6, 7 and 8 of the IPEEE submittal [1] were examined in this HFO-events review. 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 standard review plan (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, results pertaining to frequencies of occurrence of hazards, and pertaining to estimates of conditional probabilities of failure, if any, were checked for reasonableness. Review team experience was relied upon to assess the validity 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 1.1. This subsection describes the licensee's seismic evaluation in greater detail, and discusses observations of the present review.

2.1.1 Overview and Relevance of the Seismic IPEEE Process

Background

Kewaunee Nuclear Power Plant is a single-unit, 2-loop Westinghouse PWR. The plant is located in Kewaunee County, Wisconsin, along Lake Michigan's western shoreline. The plant commenced commercial operation on June 16, 1974.

The large, dry containment at Kewaunee is of Westinghouse design and consists of a primary free-standing steel containment vessel, surrounded by a reinforced-concrete shield building, with an annular space between the two structures.

The safe shutdown earthquake (SSE) for Kewaunee is characterized by a 0.12g peak ground acceleration (PGA) for horizontal motion. The plant operating basis earthquake (OBE) is 0.06g PGA for horizontal motion. The design accelerations for vertical motions are taken to be two-thirds of the corresponding values for horizontal motions. The design spectral shape is defined by a Housner spectrum. All Class I structures/rooms have been designed for seismic loads obtained from these design motions. The turbine building and spent fuel handling area have been designed to seismic requirements of the UBC (uniform building code), 1967 edition for seismic Zone 1. The auxiliary building and containment structures are founded on a common, rigid foundation mat; the turbine building and battery room are constructed on a separate rigid mat foundation. These foundations rest on clay-sand soil deposits with an approximate depth to bedrock of 76 ft.

Seismic Review Category and RLE

Kewaunee is assigned to the focused-scope seismic review category in NUREG-1407. The review level earthquake (RLE) for evaluation of the plant has been established at 0.3g PGA, with spectral shape defined by the NUREG/CR-0098 [15] median spectrum for soil conditions.

Seismic IPEEE Process

A new seismic probabilistic risk assessment (SPRA), including qualitative and quantitative (Level-2) containment performance analysis, was conducted for the seismic IPEEE. Kewaunee is a USI A-46 plant; the USI A-46 evaluation effort was coordinated, to a significant extent, with the seismic IPEEE effort. (For example, component fragilities used in the seismic IPEEE were frequently evaluated from results of USI A-46 calculations. In addition, the relay chatter evaluation for USI A-46 served as the basis for a review of bad-actor relays within the scope of the IPEEE.)

Review Findings

A well-executed seismic PRA is clearly relevant to assessing the resistance of Kewaunee to potential severe seismic accidents. The licensee's overall IPEEE process is thus judged to be consistent with the approach requested by NUREG-1407. However, it is important to note that the use of the surrogate element for Kewaunee has presented an obstacle with respect to obtaining full SPRA insights from the seismic evaluation. In particular, a meaningful list of dominant contributors has not been produced.

2.1.2 Logic Models

The plant logic analysis for Kewaunee has included the following three major aspects: (a) seismic initiating events analysis, (b) development of seismic event trees, and (c) development of seismic fault trees.

Seismic Initiating Events Analysis

Seismic initiating event categories for the Kewaunee IPEEE include: reactor vessel rupture, loss of coolant accidents (LOCAs) (large, medium, and small), transients, and plant-specific initiators. From these major categories, twelve failure groups were defined:

1.	(CSF)	- Containment or steam generator failure
2.	(RVB)	- Reactor vessel, RCS piping, or building failures
3.	(OSB)	- Offsite power failure
4.	(ACP)	- AC power failure
5.	(SWS)	- Service water system failure
6.	(DCP)	- DC power failure
7.	(RCF)	- RCS component failure
8.	(ROD)	- Control rod insertion failure
9.	(MPB)	- Medium primary pipe break
10.	(SPB)	- Small primary pipe break
11.	(SEAL)	- Reactor coolant pump seal LOCA
12.	(SSP)	- Secondary side pipe break

A seismic event tree was constructed with each of these 12 failure groups modeled as a top event in the tree. The initiating event tree logic starts with occurrence of a seismic event, which is then ultimately mapped into 12 possible end states. Of these twelve initiating event end states, six lead directly to core damage (and hence, are not modeled further), one was assumed to have zero probability of occurrence, and the remaining five define initiators that are resolved further in the seismic event tree analysis.

Seismic Event Trees

For each of the five initiators defined in the initiating events analysis, the licensee had already developed event tree logic in the IPE. This logic formed the foundation for constructing the corresponding seismic event trees. Each of the five IPE event trees were modified to account for seismic effects. The IPEEE submittal report provides a detailed description of these modifications (Section 3.1.5.2). The event tree modifications assume that instrument air and offsite power are both lost in a seismic event.

Seismic Fault Trees

Seismic fault trees were developed to model the failure logic of event tree top events and of support systems; not all top events required such modeling. Only components that were screened in during the screening process and plant walkdown were modeled rigorously in the seismic fault trees. The surrogate element was used to represent all possible seismic failures of screened-out components in a given system. The surrogate element was modeled as a basic event in series with the seismic fault tree logic for screened-in components. The resulting seismic fault trees were then linked with IPE fault trees which include non-seismic failures, human errors, and test and maintenance unavailabilities.

The IPEEE submittal report clearly describes various assumptions made in constructing the seismic fault trees (e.g., assumed failures, mission times, treatment of dependencies, etc.).

Review Findings

The submittal provides a clear and adequate explanation of plant seismic severe-accident functions. The submittal describes significant details of plant configuration, sufficient to suggest that the actual plant configuration has been represented in the IPEEE. Specific information (walkdown notes and worksheets) have been provided by the licensee, which help verify the adequacy and reasonableness of the IPEEE's modeling/treatment of plant configuration.

The logic modeling performed for the Kewaunee seismic IPEEE appears to have been well-conducted and documented, addressing all significant modeling issues in a meaningful way. The treatment of the surrogate element in the seismic plant logic analysis (i.e., as a basic event in series with remaining seismic fault tree logic) is, however, only an approximate approach. Rather than modeling failures of screened-out components as individual basic events, this approach assumes that all failures of screened-out components can be characterized by a single basic event. The resulting seismic system fault trees have limited ability to realistically represent seismic severe accident response. The approach is said in the submittal to be conservative because screened-out components will generally have capacities significantly in excess of the surrogate element capacity. In cases where more than one screened out component, in a given system, has a capacity close to the surrogate element capacity, the approach will be non-conservative; however, such cases are not likely to arise consistently in every system modeled. Since the surrogate element is modeled as a basic event in several plant systems modeled as event tree nodes, the modeling approach should tend to be somewhat conservative.

It is noted that the licensee's submittal conservatively assumes unavailability of offsite power, instrument air, and failure of the CST in any seismic event, and also makes conservative assumptions concerning failure dependencies.

Overall, the plant logic analysis is judged to be capable of producing a conservative bound on seismic risk.

2.1.3 Non-Seismic Failures and Human Actions

Because seismic fault trees were linked with IPE fault trees, non-seismic failures, human actions, and test and maintenance unavailabilities were all explicitly included in the seismic IPEEE.

Seismic impacts on operator error rates were modeled by means of fragility curves developed in the following fashion:

- For low ground motions, up to and including the SSE (0.12g), the operator error rates are the same as those used in the IPE.

- For ground motions between one to three times the SSE, the operator error rates are linearly scaled with ground motion, from a value of one times the IPE rate (at the SSE) to a value of ten times the IPE rate (at three times the SSE).

- For ground motions above three times the SSE (i.e., 0.36g), the operator error rates are set to unity.

Operator actions to reset relays were apparently not modeled in the seismic IPEEE.

The licensee has explicitly included the effects of non-seismic failures and human actions by linking seismic fault tree logic with IPE logic models that account for these effects. The licensee has implemented an approximate operator fragility method for accounting for seismic effects on human error rates. This methodology appears to be over-simplified, producing unrealistic results. The submittal does not document the location and timing associated with the required human actions. Also, recovery from relay chatter is not modeled in the seismic analysis; however, the submittal suggests that bad-actor essential relays will be replaced, or circuitry will be redesigned.

Overall, the licensee's treatment of non-seismic failures and human actions is judged to satisfy the requested guidelines of NUREG-1407. Because operator fragility functions are not considered to be very realistic, caution should be exercised in interpreting the SPRA findings, in order to ensure that operator failures do not mask insights pertaining to seismic failures of components.

2.1.4 Seismic Input (Ground Motion Hazard and Spectral Shape)

Component fragility curves were, in general, developed using the 10,000-yr median 1989 LLNL UHS spectral shape. The surrogate element fragility, however, is a special case in the sense that it is ultimately derived from screening-based spectral ordinates that show greater similarity to a NUREG/CR-0098 [15] spectral shape (even though the Kewaunee IPEEE derives the surrogate element median capacity with respect to the UHS spectral shape). In actuality, therefore, a single, consistent seismic input spectrum has not been used in the analysis. That is, fragilities (and hence, HCLPFs) of screened-in components are determined with respect to the UHS spectral shape, whereas screened-out components are represented by a surrogate element having a HCLPF capacity of 0.3g with respect to a NUREG/CR-0098 shape. The submittal reports plant HCLPFs with respect to the site-specific UHS shape. (Since the surrogate element was found to have a dominant impact on the plant-level HCLPF capacity, however, it is not clear that the UHS shape is the most appropriate for reporting the plant-level HCLPF capacity.)

Both the EPRI and 1993 LLNL mean seismic hazard curves for PGA were used in quantifying seismic initiating event frequencies and in determining seismic core damage frequency (CDF) results.

For a seismic PRA, NUREG-1407 recommends the use of the 1989 LLNL median, 10,000-yr UHS spectral shape as the basis for characterizing ground-motion input. NUREG-1407 also recommends that LLNL and EPRI mean seismic hazard curves be used for risk quantification. If the licensee chooses to use only one mean hazard curve, NUREG-1407 requests that the higher of the two be used. For a seismic margin assessment, NUREG-1407 requests that the median NUREG/CR-0098 spectrum be used to characterize seismic input.

The licensee's seismic IPEEE has substantially conformed to these recommendations, and hence, the seismic input spectrum and ground motion hazard used for the Kewaunee IPEEE are judged to be consistent with the relevant guidelines presented in NUREG-1407.

It is considered somewhat problematic that the plant-level capacities are reported with respect to the UHS spectral shape, whereas the surrogate element -- which has been identified as the dominant contributor -- has a capacity most closely related to a NUREG/CR-0098 spectrum. As a consequence, for vibration frequencies below about 1.2 Hz, the reported plant HCLPF capacity (with a PGA of 0.26g) is less than the plant's seismic design basis. Because, in fact, most components have capacities in excess of the (surrogate element) screening level, and because the screening level is itself applicable over a broad frequency range, it is believed that this result is largely artificial. (In other words, the actual plant HCLPF capacity most likely envelopes the plant seismic design basis.)

2.1.5 Structural Responses and Component Demands

New in-structure response spectra (IRS) were generated based on existing structural models. The structural models used are the original dynamic response models developed for SSE design analyses. The model of power-block structures is a three-dimensional, lumped-mass model with soil springs. This model was re-analyzed using a motion consistent with the UHS spectral shape, in order to obtain dynamic structural responses and amplified IRS for the seismic IPEEE. Structural damping for all modes was set to 7%.

For the seismic IPEEE, structural responses and in-structure response spectra should be obtained based on appropriate structural-dynamic models, using the established seismic input(s), and consistent parameters and criteria. Best-estimate structural model parameters should be used. Existing final safety analysis report (FSAR) IRS can be used/scaled to define demands, or new IRS may be generated.

In the Kewaunee IPEEE, new IRS were generated based on existing dynamic models, best-estimate damping, and PRA motion input. The development of structural responses and component demands is consistent with the relevant guidelines presented in NUREG-1407.

2.1.6 Screening Criteria

The screening criteria described in EPRI NP-6041 [5] defined the framework used in making screening decisions. The first screening column of Table 2-3 of EPRI NP-6041 was selected for the screening process. Hence, all components meeting the screening criteria are screened-out at a HCLPF level of 0.3g PGA (0.8g S_a).

The submittal also notes that the GIP [6] screening criteria were applied for evaluation of components. Simplified fragility levels were assigned to IPEEE components based on Conservative Deterministic Failure Margin (CDFM) calculations performed as part of the USI A-46 evaluation.

Nearly all components either screened out at the relatively low ground-motion screening level or have computed capacities in excess of the screening level. The use of the surrogate element to model the effects of screened-out components has effectively obscured the development of meaningful insights pertaining to real dominant risk contributors. For evaluation of a significant set of dominant contributors, the screening threshold would need to be defined at a higher level.

Although the screening approach obscures meaningful insights concerning risk contributors, it is entirely satisfactory as a seismic margin screening basis, and thus, it is judged to be appropriate in achieving NUREG-1407 objectives for Kewaunee.

2.1.7 Plant Walkdown Process

Significant coordination of seismic walkdowns was implemented to achieve the objectives of the IPEEE and of USI A-46. All IPEEE components were documented as USI A-46 items, even if they were not part of USI A-46. A Seismic Evaluation Work Sheet (SEWS) was completed for each IPEEE equipment item, in accordance with GIP requirements, and a simplified, CDFM-based fragility value was developed for each component. GIP criteria and EPRI NP-6041 walkdown procedures were followed in the walkdowns.

A number of seismic review teams (SRTs) participated in the walkdowns. Each team consisted of two seismic capability engineers trained by EPRI in USI A-46 walkdown requirements and in IPEEE add-on requirements. SRT members were drawn from WPSC staff and consulting organizations (Stevenson & Associates, Jack R. Benjamin & Associates, and RPK Structural Mechanics Consulting). The walkdowns took place over approximately a three-week time period.

In addition to walkdown of structures and active equipment, safety-related piping, electrical raceways, and ductwork were also addressed. Essential relays were evaluated based on screening rules and circuit analysis, and were spot-checked in the walkdown to confirm type, location, and installation adequacy.

A two-day walkdown peer review (with an additional one-day documentation review) was conducted by Dr. Paul Smith, and was based on GIP requirements.

The seismic IPEEE walkdowns of Kewaunee involved a significant effort by trained licensee personnel and consultants. The walkdown process is judged to have been well-executed, capable of identifying outliers with respect to anchorage, interaction, construction adequacy, and function, and has been an appropriate basis for evaluating component fragilities. Thus, the walkdown process appears to be a strong point of the study, has implemented appropriate procedures, and appears to have addressed all major items of concern.

2.1.8 Fragility Analysis

Three type of fragility analyses were employed in the Kewaunee seismic IPEEE:

1. Surrogate element fragility assessment

2. Simplified fragility assessment
3. Detailed fragility assessment

The surrogate element fragility is used to describe the seismic capacity of all components in a given system that have been screened out. The median PGA capacity of the surrogate element is obtained by enveloping the UHS spectral shape by the screening spectrum converted to a median capacity. A composite logarithmic standard deviation of 0.3 is used to define the surrogate element. A PGA value of 0.64g was determined as the surrogate-element median capacity. In performing the enveloping, the licensee did not properly extrapolate the UHS over the high frequency range. Had a proper extrapolation approach been used, a somewhat lower median capacity would have been assessed.

Simplified fragility assessment was applied to most screened-in components. Simplified fragilities (median capacities) were generally obtained from results of CDFM HCLPF evaluations. A composite logarithmic standard deviation of 0.46 was used for all simplified fragilities.

Detailed fragility analysis was essentially applied to only one screened-in component, i.e., RHR heat exchangers. The methodology employed for detailed fragility assessment was the conventional approach based on median safety factors and derivation of combined variability from elemental safety-factor variabilities. The RHR heat exchangers were determined to have a median capacity of 0.63g PGA and a composite logarithmic standard deviation of 0.46. Median capacities determined from detailed fragility assessment of other base-mounted equipment were found to all exceed the surrogate element median capacity.

The approach implemented for component fragility evaluation in the Kewaunee IPEEE is considered to be well-structured and meaningful. The use of simplified and detailed approaches to fragility assessment is a valid and reasonable basis for analysis. The use of a surrogate element fragility function for screened-out components is also, in principal, considered to be well-conceived and appropriate, provided that the screening threshold is chosen at a sufficiently high level. The licensee's selection of a low screening threshold, however, has produced a surrogate element fragility that limits the ability to identify meaningful risk contributors.

2.1.9 Accident Frequency Estimates

Quantification of seismic accident sequences was performed using the Jack R. Benjamin & Associates SHIP code. In this quantification, the seismic hazard curve is discretized to obtain initiating event frequencies for various ground motion levels. For each ground motion level, component fragilities are evaluated to obtain basic event probabilities in the seismic fault trees. Random failure probabilities (from IPE data) and operator error probabilities (derived from operator fragility curves) are obtained to quantify the IPE portion of the systems fault tree logic. Probabilities determined from system fault trees define event tree top event failure/success probabilities. The SHIP code evaluates a fragility curve for each top event. Event tree accident sequence logic is used to quantify accident sequence frequencies from top event probabilities. The SHIP code evaluates a sequence-level fragility curve for each end node of the event trees. In addition, a core damage fragility curve is obtained by combining sequence-level fragility curves. The core damage frequency is obtained by appropriately combining the seismic hazard curves and plant-level fragility curves.

The Kewaunee seismic IPEEE submittal reports frequencies for the 47 accident sequences modeled in event trees. The submittal also presents the system-level, sequence-level, and plant-level fragility curves. CDFs are reported for both EPRI and 1993 LLNL seismic hazard curves.

The approach for the licensee's assessment of accident sequence frequencies is clear, accurate and well-executed. The computer code used to develop accident frequency estimates has been subjected to quality assurance procedures. The actual frequency estimates are believed to be somewhat conservative due to the use of the surrogate element and due to assumptions made in approximate treatment of the surrogate element in systems fault tree logic. The presentation of system-level, sequence-level, and plant-level fragility curves are viewed to be a significant strength of the study. In addition, the submittal provides a clear presentation of dominant accident sequences and a table of accident sequence frequencies.

2.1.10 Evaluation of Dominant Risk Contributors

Dominant basic events/component failures that contribute to seismic risk were determined based on their contribution to plant fragility. Dominant accident sequences and plant systems were also similarly determined in the IPEEE report. The process for evaluating the dominant basic events, sequences, and systems is not explained in detail in the IPEEE submittal.

Dominant Contributors to Core Damage

The seismic IPEEE submittal has identified the following dominant risk contributors to core damage frequency: offsite power, the surrogate component (used in modeling various systems), and operator error (failure to switch auxiliary feedwater (AFW) pump supply from condensate storage tank [CST] to SW). This list of dominant contributors is not considered to be very meaningful; i.e., use of the surrogate component acts to hide understanding of the true dominant risk contributors.

Dominant Contributors to Radioactive Release given Core Damage

Other than those items already identified as core damage dominant risk contributors, the seismic IPEEE does not specifically list additional dominant contributors to radioactive release given core damage. A Level-2 analysis was apparently performed for assessing seismic containment performance. No formal analysis was undertaken to find vulnerabilities in containment safeguard (CSG) systems, including containment isolation (SXCI), containment air cooling (SXFCH) and containment spray (SXICS). Among the CSG systems, SXICS has the lowest capacity (fragility); the shape of the fragility function reveals that operator error dominates failure of this system. Hence, operator error is modeled as having a substantial impact on containment performance at Kewaunee. Due to use of the surrogate element, actual components having a dominant contribution to seismic containment failure risk are unknown. However, all components in the CSG systems, like those required for accident prevention, were found to have HCLPF capacities no lower than about 0.3g PGA. (Again, this HCLPF capacity is reported with respect to a site-specific spectral shape, not the NUREG/CR-0098 shape.)

Review Findings

The Kewaunee seismic IPEEE does not produce valid insights with respect to dominant risk contributors. The study's finding that the surrogate element is a dominant risk contributor is an artificial result that lacks real meaning with respect to plant behavior.

2.1.11 Relay Chatter Evaluation

A relay chatter evaluation was conducted as part of plant assessment for USI A-46. This review encountered 12 instances of bad-actor relays at Kewaunee (all Westinghouse, Model SC relays). Consequently, the scope of the bad-actor relay review was expanded to include IPEEE equipment which were not also part of the scope of USI A-46. No additional bad-actor relays were identified.

The submittal notes that the 12 bad-actor relays are either to be replaced, or their circuitry is to be re-worked.

The licensee's evaluation of relay chatter for Kewaunee appears reasonable and consistent with NUREG-1407 guidelines.

2.1.12 Soil Failure Analysis

The Kewaunee IPEEE submittal includes analyses of the following three categories of potential soil failures:

1. Liquefaction
2. Transient and permanent displacements and settlements of buildings
3. Displacements of buried piping from the screenhouse to the intake crib

A description of site soils and their dynamic characteristics is provided in the submittal.

The submittal reports that the potential for soil liquefaction beneath power-block structures or the screenhouse structure is very unlikely. The submittal reports computed maximum transient and permanent displacements and settlements. These settlements are used to conduct a fragility analysis for buried piping connecting the screenhouse to the intake crib. Based on the analysis, buried piping was screened out at a PGA level of 0.7g, and its fragility was subsequently represented by the surrogate element.

The treatment of soil failures in the Kewaunee seismic IPEEE is judged to satisfy the guidelines described in NUREG-1407 for a focused-scope plant.

2.1.13 Containment Performance Analysis

The containment performance analysis developed seismic fragility curves (based on seismic fault tree logic) for the following containment safeguard systems:

- Internal containment spray
- Containment air cooling
- Containment isolation

A plant walkdown was conducted of containment systems, as well as the containment structure itself. The containment structure (including penetrations, hatches, isolation valves, concrete wall, steel shell, piping, and conduit) was found to meet the screening criteria, and was thus represented by the surrogate element.

The Kewaunee relay chatter evaluation included consideration of relays associated with the actuation signals for the CSG systems and the ECCS systems.

A Level-2 analysis was conducted to quantify release category frequencies. In this analysis, only accident sequences (from the Level-1 analysis) having a CDF contribution greater than 10^{-7} per reactor-year (ry) were included in the Level-2 quantification. These sequences account for over 96% of the total seismic CDF. The IPE containment modeling was assumed to be applicable for the seismic analysis, except for seismic sequence SCSF, which involves a catastrophic containment failure leading to release of over 10% of volatile fission products.

A seismic containment failure frequency, and containment failure fragility and HCLPF capacities, were estimated based on the Level-2 seismic analysis. The submittal also reports release categories and their seismic-induced occurrence frequencies.

The seismic IPEEE submittal does not identify any containment performance vulnerabilities, other than items noted during walkdowns. The relatively low HCLPF capacity of 0.3g associated with containment failure (Release Category U or G) leads to questions concerning the level of conservatism introduced in the analysis or the need to enhance the seismic capability of containment performance. The use of the surrogate element has clearly led to conservatisms in the containment performance assessment. However, operator actions have also been identified as having an important effect on containment performance. It is considered prudent for the licensee to consider procedural enhancements which may increase relevant operator reliability; however, more realistic assessment of operator fragilities should be addressed in such consideration.

Aside from these problems, the general Level-2 approach to containment analysis, implemented for the containment performance evaluation, is itself clear and detailed, and exceeds NUREG-1407 guidelines. The qualitative assessment is judged to be valid and meaningful. The quantitative assessment is judged to have produced a conservative estimate of containment capacity and failure frequency; however, the assessment has not produce reliable insights concerning dominant contributors to early seismic containment failure.

2.1.14 Seismic-Fire Interaction and Seismically Induced Flood Evaluations

Section 4.8.5 of the Kewaunee IPEEE documents an analysis of seismic-fire interactions, which includes the following aspects:

- Seismic-induced fires
- Seismic degradation of fire suppression capabilities
- Inadvertent actuation of fire suppression systems

For seismic-induced fire considerations, the IPEEE evaluated the following items: pumps containing lube oil, the turbine lube oil storage tank, diesel generator (DG) fuel oil day tanks, and gas bottles. The submittal concluded that seismic-induced fires are not a credible threat at Kewaunee.

For consideration of seismic effects on fire suppression capability, the IPEEE evaluated the fire water system, the CO_2 system, and the DG Cardox system. It was found that a seismic event could damage fire water capability, but that damage to the CO_2 system is not credible; also seismic-induced actuation of the

DG Cardox system was concluded not to present a hazard. Mercury switches were found in the seismic walkdowns, including A and B fire pump jockey switches and Cardox pressure switches. It was evaluated that the worst consequence of failure of any of these switches would be an unavailability of fire protection systems.

The submittal notes that charged fire water sprinkler systems are not located in safety-critical areas, and hence, the impact of inadvertent actuation is minimized.

The Kewaunee IPEEE has implemented a seismic-fire interactions evaluation which is reasonably complete.

2.1.15 Treatment of USI A-45

The Kewaunee submittal provides a detailed discussion of USI A-45, "Shutdown Decay Heat Removal Requirements." For seismic events, discussions are provided in the submittal relative to secondary cooling, bleed and feed cooling, and ECCS injection and recirculation.

The main feedwater (MFW) system is assumed to be unavailable due to failure of offsite power. Bleed and feed cooling is conservatively assumed to be unavailable due to loss of instrument air (needed to operate pressurizer PORVs). The auxiliary feedwater (AFW) system and the ECCS (safety injection and residual heat removal) are thus the only forms of decay heat removal (DHR) credited in the seismic evaluation. AFW capability requires successful DG operation and operator action to switch AFW pump suction from the CST to the service water (SW) supply.

The only significant finding with respect to DHR capability is the seismic capacity (median of 0.63g PGA) of the RHR heat exchangers, which controls the capacity of the ECCS.

The IPEEE submittal concludes that, because several failures would be needed to impact DHR capability, and because the computed CDF is low, the licensee's requirements with respect to USI A-45 are fulfilled.

The seismic IPEEE submittal for Kewaunee contains a meaningful discussion on the shutdown decay heat removal capability of the plant (for USI A-45 resolution). Substantial detail is provided that clearly explains the specific features of the plant in this regard. These plant features appear to have been adequately and appropriately modeled in the SPRA. Hence, conclusions drawn in the seismic IPEEE should be appropriately relevant to shutdown decay heat removal capability. The conclusions suggest that there are no seismic-related vulnerabilities with respect to decay heat removal at Kewaunee for a site-specific RLE.

Thus, the Kewaunee seismic IPEEE includes a meaningful evaluation of potential vulnerabilities in decay heat removal systems, which is judged to address the relevant concerns of USI A-45.

2.1.16 Treatment of GI-131

The Kewaunee seismic IPEEE submittal includes a brief discussion on GI-131, which pertains to movable in-core flux mapping systems in Westinghouse plants. This issue is not directly applicable to Kewaunee because the flux mapping cart is not movable. However, the seismic resistance of the stationary ten-path flux mapping frame was investigated; a dynamic analysis of the mapping frame was conducted for this the purpose. The dynamic analysis showed that the mapping frame could easily sustain seismic forces without the aid of lateral restraints.

A walkdown was performed to examine potential interaction problems. The walkdown revealed a potential interaction from a chain-fall on a hoist attached to an I-beam above the mapping cart. The I-beam is cantilevered from a nearby concrete wall. Apparently, the system is sufficiently flexible that the chain fall might impact the flux mapping cart table. In response to this potential adverse interaction, the licensee has implemented administrative procedures to help ensure that the hoist is restrained at the fixed end of the I-beam/crane rail when not in use. GI-131 is thus considered resolved by the licensee.

The Kewaunee IPEEE includes an apparently meaningful evaluation of concerns related to GI-131.

2.1.17 Other Safety Issues

USI A-46, USI A-17 and USI A-40 Resolution

A significant effort in coordination of USI A-46 and the seismic IPEEE has taken place for the seismic evaluation of Kewaunee. USI A-46 is resolved separately from the seismic IPEEE. The submittal notes that resolution of USI A-17 and USI A-40 will be addressed in the USI A-46 submittal. Hence, this TER does not include an evaluation of the licensee's treatment of these issues.

Eastern U.S. Seismicity Issue

As a result of probabilistic seismic hazard analyses performed for Eastern U.S. plant sites, five plants were identified as outlier sites. NUREG-1407 states that the IPEEE will provide a resolution for the outlier plants with no need for additional analyses or documentation from licensees. The Eastern U.S. seismicity issue is known also as the Charleston Earthquake Issue.

Probabilistic seismic hazard calculations were performed for the Kewaunee site, as part of the resolution program for the Charleston Earthquake Issue. Kewaunee was not identified as an outlier plant.

Review Findings

The seismic IPEEE includes discussions concerning USI A-46, USI A-17, USI A-40, and the Eastern U.S. Seismicity Issue; these issues are not considered further in this review.

2.1.18 Process to Identify, Eliminate, or Reduce Vulnerabilities

The Kewaunee seismic IPEEE submittal has identified no plant vulnerabilities, and hence, has not proposed specific actions to eliminate or reduce vulnerabilities. No definition of vulnerability, nor systematic process to identify vulnerabilities, was documented in the submittal report.

The submittal does report a number of outliers that have required resolution approaches, including plant safety enhancements. These safety enhancements are significant, and are described in Table 3-4 of the IPEEE report (repeated in this TER as Table 4.1).

The identification of physically evident seismic deficiencies in the plant walkdowns is considered to be generally well-executed. The licensee's evaluation and treatment of safety enhancements to eliminate or reduce the effects of these deficiencies is generally clear. However, it is not apparent that these enhancements always address concerns beyond the design basis earthquake (DBE) level, nor that any of these enhancements are outside the scope of USI A-46 (i.e., for IPEEE-only considerations).

2.1.19 Peer Review Process

The IPEEE for Kewaunee has included an independent external peer review of the seismic walkdown process, including a review of relevant seismic documentation packages. No other external peer review of the seismic analysis is described in the submittal. An independent internal peer review of the seismic analysis was conducted by WPSC engineers and middle managers. The submittal notes that all areas of the IPEEE were subject to review, and that all reviewer comments were formally documented and resolved.

In conclusion, a peer review, consistent with NUREG-1407 guidelines, was conducted as part of the Kewaunee seismic IPEEE.

2.2 <u>Fire</u>

A summary of the licensee's fire IPEEE process has been described in Section 1.2 of this TER. 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. Method Selected for Fire IPEEE

The fire hazard is analyzed in two major steps. In the first step, screening is done based on the FIVE methodology. In the second step, PRA methodology is used for the surviving locations. PRA methodology is employed for specific locations within the unscreened fire areas.

b. Key Assumptions Used in Performing Fire IPEEE

A list of assumptions are provided on Pages 4-10 and 4-11 of Reference [1]. The key assumptions, with respect to significant effect on results, are:

1. Fire barriers/boundaries are good as rated. Active systems (for example self closing/normally open fire doors) are part of fire barrier definition. Some consideration is given to the possibility of open doors, ducts, failure of fire dampers, etc. This results in cross-zone fires being judged to have negligible risk.

2. The design of the automatic fire suppression systems are assumed to be perfect. That is, if detection occurs, suppression actuates instantly and the fire will always be put out. It is stated that "all automatic fire suppression systems are sized to effectively mitigate maximum sized fire". This could be an optimistic assumption. The information provided in Reference [1] is not adequate to properly examine the adequacy of this assumption.

3. All fires in zones containing safe shutdown equipment lead to reactor trip.

4. In-cabinet fires do not spread to other cabinets.

c. Status of Appendix R Modifications

Appendix R modifications are assumed to be completed. An audit was conducted in 1991.

d. New or Existing PRA

The IPEEE is a new PRA, and not based on an existing PRA.

2.2.2 Review of Plant Information and Walkdown

a. Walkdown Team Composition

Two Westinghouse engineers and four WPSC engineers conducted the fire walkdown. The WPSC personnel individual areas of expertise were as follows: risk assessment, nuclear engineering, fire protection operations supervisor, and quality assurance auditor.

b. Significant Walkdown Findings

The entire plant, except for the containment building, was inspected. The fire zones that required detailed analysis were examined closely. As part of the walkdown, the physical conditions of active fire barriers were examined. The team has examined such issues as whether fire doors are blocked open and "fusible links were examined and determined to be in good repair". It is difficult to envision how the team could ascertain the conditions of the fusible links outside examining the stamped temperature rating, and checking whether rust or dirt had developed around the link. The team could identify obstructions to fire dampers and roll-up doors. The licensee may have reached an optimistic conclusion regarding fire vulnerabilities by the qualitative screening of fire propagation across fire zones. Active fire dampers have displayed a poor reliability (e.g., failure probability can be as high as 0.20 per demand).

In the case of screenhouse fire events, credit is taken for the presence of large fans that would exhaust fumes and hot gases from these fire zones. However, there are no discussions as to whether the fans are powered by cables and electrical cabinets from outside the zone where the fire is postulated.

c. Significant Plant Features

The following is a list of plant features that are deemed to be important:

1. Reactor coolant pump oil collection system
2. AFW system not dependent on external cooling
3. AFW pumps trip on low discharge pressure
4. Charging pumps not dependent on external cooling
5. Important cabling runs through and is accessible to fires in the diesel generator rooms
6. High pressure SI pumps capable of pumping against 2200 psi
7. Eight hour rated station batteries

8. Diesel Generator Room B contains 4160 V cabinets and cable routing from numerous safe shutdown-related systems
9. AFW Pump Room A contains 480 V electrical buses and cable routing from numerous safe shutdown-related systems
10. AFW Pump Room B contains Dedicated Analog Controller and cable routing from numerous safe shutdown-related systems

2.2.3 Fire-Induced Initiating Events

a. Were Initiating Events Other than Reactor Trip Considered?

A separate discussion is provided for this subject (Section 4.1.3 of Reference [1]). A comprehensive list of initiating events is addressed. Special attention is given to the possibility of hot shorts leading to a valve opening inadvertently. However, in Reference [13], it is stated that for control room fire analysis it is assumed that a fire would cause the fuse of affected control circuits to blow, and the equipment to fail in their normal mode. Although, in several instances, it can be inferred that the licensee had considered the possibility of a hot short affecting the control circuits (e.g., the possibility of loss of offsite power), the assumption regarding the protection provided by fuses in case of a control panel fire may not be valid. Wire to wire contacts that simulate the effect of a switch on the control panel can be envisioned. Such contacts may be the result of insulation failure in a fire, and will not cause an abnormal current within the circuit, and therefore, will not blow the fuse. From the discussions provided in Reference [13], it can be inferred that optimistic assumptions have been employed in analyzing the possibility of a small LOCA from spurious opening of a PORV, and inadvertent steam dump from spurious opening of a steam dump valve. The licensee concludes that the valves would reclose upon loss of power from a blown fuse. The submittal also states that the procedure for using the remote shutdown panel requires the fuses of the control circuit to be removed prior to switchover to the remote shutdown panel.

b. Were the Initiating Events Analyzed Properly?

1. Some discussion is provided as to which initiating events are considered as possible to occur (from a fire event). Loss of offsite power is used in the initial screening of fire areas.

2. For the possibility of a PORV opening from fire outside the containment, it can be inferred that the licensee has traced the proper cables for this event, and has identified the locations where failure of these cables could lead to a small LOCA. It is claimed that PORV fuses will be removed to assure closure. The possibility of PORV failure to reseat, and thus occurrence of a small LOCA, has not been quantitatively evaluated.

2.2.4 Screening of Fire Zones

a. Was a Proper Screening Methodology Employed?

The screening methodology starts at a high level (i.e., whether any safety related cables or equipment are present in a zone) and gradually builds more information into the analysis until a fire propagation and suppression analysis is done. The fire scenarios have been screened out based on 10^{-6} core damage frequency (per reactor-year).

The loss of offsite power event tree is used for establishing the frequency of core damage. The human error rates are not altered from those used in the IPE. From a sensitivity analysis conducted by the licensee, it is concluded that human error rates are not an important contributor to the screening results and do not affect the relative ranking of the fire areas (in terms of risk significance).

Given the overall results of the analysis, this screening level is considered adequate.

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

A conservative approach is used for the cable spreading room. The fire occurrence frequency is small compared to the industry norm. It is conservatively assumed that a fire will destroy all the cables in the room. The possibility of operator response and use of alternate shutdown methods are not analyzed.

A thorough analysis of the control room has been conducted. It is stated that in some cases the switches for redundant trains are 6" apart. However, for these switches there are small metal plates separating them. The probability of failure to suppress the fire before damage is taken to be 10^{-4} per occurrence. This value is certainly much smaller than what is typically used for fire suppression failure probability. As discussed in Section 2.2.3, the possibilities of small LOCA, inadvertent steam dump and loss of offsite power have been considered for control room fires. However, the underlying assumption regarding blown fuses protecting the affected circuits may have led to optimistic results.

The licensee has conducted an investigation of every control cabinet and section of control console. It is assumed that fire will not propagate to other panels or sections of the panel. From this analysis it is concluded that a fire in Electrical Console A may lead to core damage by failing the breakers of a vital bus. The core damage frequencies for the two scenarios for this console are concluded to be 1.4×10^{-5} and 1.8×10^{-5} per reactor-year. These frequencies are certainly larger than that concluded for other power plants.

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

The justifications provided for all the fire zones are reasonable. In some cases, combustible loading and fire protection features of the fire zone are used as the sole basis for screening out the area from further analysis. This cannot be an acceptable approach if an area contains vital redundant trains. From the information provided by the licensee, the reviewers have concluded the screening results are within a reasonable range.

In particular, regarding Fire Zone TC-100, it should be noted that, per Reference 22, the area contains one train of a large number of vital systems. Based on this information, the reviewers can concur with the licensee that the risk of a fire in this fire zone may not be significant.

2.2.5 Fire Hazard Analysis

The FIVE database and initiation frequency methodology were used for establishing the frequency of fire in an area. A plant-specific database has not been used. This could be somewhat non-conservative, since the fire database shows that, at Kewaunee, a large fire has occurred in the main auxiliary transformer bay, and another fire has occurred in a diesel generator room. The licensee has stated that the plant-specific database is insufficient to use.

2.2.6 Fire Growth and Propagation

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

Cross-zone fire spread is judged to be negligible because all barriers are assumed to perform as rated. The possibility of failure of active fire barriers (e.g., rollup doors) has been considered as very unlikely based on the inspections conducted during the walkdown [12]. This conclusion may be optimistic because active fire dampers can have an unreliability level as high as 0.2.

b. Assumptions Associated with Detection and Suppression

1. Perfect design of automatic fire suppression systems is assumed (i.e., if detection occurs, suppression actuates instantly and the fire will always be put out).

2. Unavailability of detection and suppression systems is considered.

3. Perfect design of fire detection (i.e., unless the detector fails, it will always detect the fire when its detection criterion is reached).

4. Manual fire detection is two hours, except in the control room. The control room is assumed to be always occupied.

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

Suppression-induced damage is not treated in either the PRA or screening analyses.

d. Computer Code Used, if Applicable

COMPBRN IIIe is used for fire damage assessment. Westinghouse's WALT program is used for estimating the core damage frequency for fire screening analysis based on IPE cut sets and WLINK for event tree/fault tree analysis of unscreened locations.

Only pilot fires external to cabinets are used. The pilot fire is taken to be 3 kg of heptane in nearly all cases.

2.2.7 Evaluation of Component Fragilities and Failure Modes

a. Definition of Fire-Induced Failures

Loss of function of equipment associated with damaged cables or damaged motor control centers (MCCs) is assumed to take place. Hot shorts are also considered for such valves as PORVs.

In-cabinet and transformer fires are not analyzed. This is a non-conservative assumption. A fire database available to the review team shows that a fire has occurred at Kewaunee, owing to a bus fault in the main auxiliary transformer bay.

b. Method Used to Determine Component Capacities

Cables are assumed to fail when COMPBRN calculates a hot gas layer temperature in excess of the cable damage criterion of 500K. No criterion is given for MCC damage. The sensitivity analysis feature and the range of cable damage temperatures provided with COMPBRN are used in fire propagation analysis.

c. Generic Fragilities Used

The cables are IEEE 383 qualified. The generic polyethylene insulation cable damage criterion has been used.

d. Plant-Specific Fragilities Used

No plant-specific failure fragilities have been used.

e. Technique Used to Treat Operator Recovery Actions

The licensee has calculated human error and recovery probabilities using the same technique as for the IPE, but has increased the stress factors to account for the fire conditions. The licensee has conducted a sensitivity analysis on the effect of the human error rates to the final conclusions. From that analysis it has been concluded that human error rates have little impact on the screening analysis results, and the relative ranking of the important fire scenarios is not strongly dependent on the human error rates.

2.2.8 Fire Detection and Suppression

Fire initiation frequencies are multiplied by fire detection and suppression probabilities, if the time of detection and suppression is shorter than the damage time. Eight fire locations survived the screening process and were treated using PRA type fire propagation analysis. In none of the eight locations was the multiplier applied, because in all cases the damage time was shorter than the detection and suppression time.

2.2.9 Analysis of Plant Systems and Sequences

a. Key Assumptions Including Success Criteria and Associated Bases

The success criteria are directly taken from the IPE.

b. Event Trees (Functional or Systemic)

The following systemic event trees were adopted from the IPE:

- Loss of Offsite Power
- Transient Without Main Feedwater
- Transient With Main Feedwater.

Core damage timing has been provided [12].

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

No dependency matrix has been provided.

d. *Plant-Unique System Dependencies*

There are no plant-unique system dependencies.

e. *Most Significant Human Actions*

1. Actions associated with Procedure E-0-06, which requires evacuation of the control room and activation of the Dedicated Shutdown Panel (DSP). As part of this procedure, it can be inferred that the operators may cause a self-induced station blackout. Self-induced station blackout is not discussed or analyzed in the submittal.

2. Actions associated with Procedure E-0-07, which deals with a fire initiated in dedicated safe shutdown zones (a plant trip is also required). As part of this procedure, similar to E-0-06, it can be inferred that the operators may cause a self-induced station blackout. This is especially the case if Train A related equipment is affected. As mentioned above, self-induced station blackout is not discussed or analyzed in the submittal.

3. Initiation of bleed and feed

4. Manual start of a diesel generator

5. Isolation of nonessential equipment and restore power to electrical bus 5

6. Establishing Service Water from DSP

7. Establishing component cooling water (CCW) to reactor coolant pump thermal barrier

8. Manually establishing AFW

9. Establishing SI

10. Establishing instrument air

2.2.10 Fire Scenarios and Core Damage Frequency Evaluation

Overall, the licensee has properly demonstrated and summarized how the core damage frequency is estimated for each fire scenario.

2.2.11 Analysis of Containment Performance

a. Significant Containment Performance Insights

Containment fires are concluded to be insignificant for Kewaunee. This appears to be based on the fact that a large fraction of containment fires are from reactor coolant pump oil fires, and Kewaunee is equipped with an oil collection system.

All fire-induced containment failures are associated with failure of isolation. The probability of isolation failure is given as 31% of the fire-induced core damage frequency. New containment failure modes associated with fires, and not previously identified in the IPE, were not identified.

b. Plant-Unique Phenomenology Considered

The same phenomenology is used as that in the Level-2 IPE analysis. Fire sequences and associated failed equipment were analyzed using the IPE containment event trees.

2.2.12 Treatment of Fire Risk Scoping Study Issues

a. Assumptions Used to Address Fire Risk Scoping Study Issues

1. Fire barriers are assumed to be qualified per the Appendix R effort.

2. Only pumps and storage tanks are addressed for seismically-induced fires. No discussion is provided on the possibility of hydrogen line break and flammable gas release.

3. Class I fire suppression equipment are considered as not susceptible to seismic degradation.

b. Significant Findings

1. No dependencies have been identified between the control room and the remote shutdown panel, or between the dedicated and alternate shutdown fire zones. However, there is no mention of the existence of any switchover or isolation switches to isolate the circuits from the control room.

2. The licensee states that there is insufficient data to analyze potential halon and CO_2 damage associated with fire suppression. However, the licensee claims that it has checked for CO_2 and halon impact scenarios during the fire walkdown. The water damage issue is also addressed. Overall it is concluded that suppression system damage is not a significant issue.

3. The licensee has conducted studies of its fire brigade response. The licensee has found that the fire brigade reaches anywhere in the turbine or auxiliary building in less than or equal to 6.4 minutes, based on fire brigade drills from 1988-1991. However, it should be noted that the fire propagation scenarios analyzed in the fire risk analysis portion of the IPEEE are found to be shorter than this value.

4. The issue of seismically-induced fires is analyzed in some detail. It is concluded that seismically-induced fires, in safety-related areas, are unlikely.

5. The suppression systems, in safety-related areas, are properly anchored to withstand a seismic event. Therefore, seismically-induced degradation of fire equipment is not a concern.

6. Inadvertent actuation of fire suppression is not a concern, because charged systems are not located in safety-critical areas.

7. The fire brigade training was surveyed as part of the Fire Risk Scoping Study and found to be comprehensive.

8. Operators are trained in conducting fire-related procedures.

9. Potential adverse effects on plant equipment by combustion products have not been addressed.

10. Barrier failures are analyzed based on combustible loading of an area. No consideration is given to the possibility of mechanical failure of active barriers (e.g., rollup doors).

2.2.13 USI A-45 Issue

a. Methods of Removing Decay Heat

The AFW system, Long Term Recirculation Mode of RHR, and Bleed and Feed are the methods considered for heat removal during and after a fire event.

b. Ability of the Plant to Feed and Bleed

The plant has this capability.

c. Credit Taken for Feed and Bleed

Credit is taken when fire does not disable power to PORVs or both SI trains. Human error and failure of DSP are also considered as part of the inability to successfully initiate feed and bleed.

d. Presence of Thermo-Lag

The licensee claims that Thermo-Lag has not been used at Kewaunee.

2.3 HFO Events

The submittal reports that the HFO-induced CDF is estimated at a screening level of 10^{-6} (Section 1.4 of Reference [1]), which is 0.5% of the plant's total CDF from both internal and external events (Figure 1-1 of Reference [1]). The HFO events which are explicitly examined include "High Winds and Tornadoes", "External Flooding", "Transportation and Nearby Facility Accidents", and "Hazardous Materials". The submittal concludes that no vulnerabilities are identified that require detailed quantification of any accident sequence (Page 1-12 of Reference [1]).

The general methodology utilized in the study follows that presented in NUREG-1407 for the analysis of other external events, which includes the following major steps:

1. Establishing a List of Plant-Specific Other External Events,

2. Progressive Screening, and

3. Documentation.

Guidelines provided in GL 88-20, Supplement 4 [2], NUREG-1407 [3], NUREG/CR-2300 [16], and NUREG/CR-5042 [17] are referenced as the basis for completion of Step 1.

Progressive screening included the following stages:

- Review of plant-specific hazard data and licensing bases.

- Identification of significant changes since the Operating License (OL) was issued.

- Establishing whether the plant and facilities design comply with the 1975 Standard Review Plan (SRP) criteria.

- Determining whether the hazard frequency is acceptably low.

- Performing a bounding analysis, if necessary.

- Performing a PRA, if necessary.

In Section 5.0.4 of the submittal, a determination is made as to which one of the HFO events needs to be analyzed. Table 5-1 of the submittal presents the results of this determination. The following subsections provide a summary of the analysis performed for each hazard.

2.3.1 High Winds and Tornadoes

2.3.1.1 General Methodology

Kewaunee has facilities that were designed and built prior to the NRC's current criteria. Thus, the NUREG/CR-5042 approach has been used for a systematic examination of the plant. In this approach, the expected frequency of exceedance of various wind speeds is assessed first. Then, the likelihood of damage to specific plant structures and components as a result of wind-induced stress is evaluated.

The proposed steps to be performed consist of the following:

- Wind Frequency Analysis
- Fragility Analysis
- Plant/Systems Analysis
- Core Damage Quantification

2.3.1.2 Plant-Specific Hazard Data and Licensing Basis

Site-specific data for the years 1887 through 1979 is obtained from NUREG/CR-2890 [18], and for the years 1980 through 1988 is obtained from the National Oceanic and Atmospheric Administration. All data is obtained from the weather station at Green Bay. The justification provided in NUREG/CR-4458 [19] for the applicability of the Green Bay data to the Point Peach Nuclear Plant is cited to be pertinent here, based on the relative locations of Kewaunee and Point Peach with respect to Green Bay, and the terrain between Kewaunee and Green Bay.

Using the family of mean wind hazard curves for straight winds developed in NUREG/CR-4458, and a design basis wind speed of 300 mph, the frequency of exceedance is determined to be insignificant (i.e., less than 10^{-6} per year) and no further analysis is performed.

To assess the tornado-induced risk, the analysis utilizes Kewaunee's Updated Safety Analysis Report (USAR) [20] data, pertaining to the tornado occurrences, to estimate the tornado path area and number of tornadoes. Using this information in combination with a model proposed by Thom [21], the mean yearly probability of a tornado striking a point in close vicinity of the site is calculated to be 4.86×10^{-4}. However, it is not clear if this value is used later in the analysis. The analysis then attempts to evaluate the frequency of tornadoes with the potential to cause damage (i.e., implying that the above frequency pertains to the occurrence of any tornado). Next, using the mean values of wind speeds and frequencies of exceedance based on WASH-1300 [22], the frequency of tornadoes with wind velocity exceeding the design basis wind speed is determined to be insignificant (i.e., less than 10^{-6} per year).

No discussion of the potential hazard posed by tornado-generated missiles is provided in the submittal. However, the tornado missile analysis documented in the plant USAR concludes that missile impact load is unlikely to cause damage to Class I structures according to the applied design criteria. No discussion is provided as to whether or not there are non-Class I structures of importance to the IPE conclusions which may not be protected against tornado missiles.

The submittal goes to some length in order to justify that the plant's design wind speed of 300 mph is conservative. However, based on the tornado wind speed values presented on Page 5-22, it is not evident that the Kewaunee design wind speed is necessarily conservative. On the other hand, the submittal does not specify whether there are any structures that do not have the design wind load capacity of 300 mph (e.g., water storage tanks or transformers), and how these structures are treated in the analysis.

2.3.1.3 Significant Changes Since Issuance of the Operating License

The submittal does not identify any significant changes since the time the plant operating license (OL) was issued.

2.3.1.4 Significant Findings and Plant-Unique Features

No significant findings are cited in the submittal. A summary of the walkdown procedures used by the licensee and the qualification of the team members performing the walkdown are not provided in the submittal.

2.3.1.5 Hazard Analysis

NUREG/CR-4458 and WASH-1300 hazard curves are used for the evaluation of hazard frequency for High Wind and Tornado hazards, respectively. In both cases, the frequency of hazard exceeding the design basis is estimated to be less the 10^{-6} per year.

2.3.2 External Flooding

2.3.2.1 General Methodology

The methodology consists of first determining the credible flooding sources. For those found credible, the plant's minimum flooding ingress level(s) (i.e., minimum levels for the flood propagation pathways to the plant) and the maximum possible external flooding levels are determined. If the plant elevation precludes flooding from these maximum flooding levels, the analysis is complete, otherwise further analysis is performed.

2.3.2.2 Plant-Specific Hazard Data and Licensing Basis

NUREG-0965 [23] is used to screen out the credibility of on-site or off-site dams as potential flooding sources, and local topography, as presented in the plant's USAR, is used to preclude flooding from the landward side of the site. Thus, Lake Michigan and intense precipitation are considered as the only credible sources of external flooding.

U.S. Geological Survey figures from the Kewaunee USAR, pertaining to Lake Michigan, are used to evaluate the Lake's potential for causing flooding at the site. The following data pertaining to the lake are reported:

- The normal water datum of 577.5 feet above Mean Sea Level (MSL)
- The lowest recorded Level of 575.4 feet above MSL
- The highest recorded level of 582.5 feet above MSL

The two screenhouse doors, at 586 feet above MSL, are identified as the lowest flood propagation pathways for the plant. The screenhouse doors are assumed to remain closed even if the lake's level were to reach to the doors' level. The licensee provides a description of the construction of the screen house doors to support the validity of this assumption.

Referring to the Kewaunee and D. C. Cook FSARs, the submittal states that no water level increase of as much as 8 feet should ever be experienced at the plant. From review of the submittal, it is not clear whether this is a licensing basis for the plant or not.

With regard to GI-103, "Design for Probable Maximum Precipitation (PMP)," the submittal appears to evaluate PMP-induced flooding risk based on the assumption that the only potential damage inducing mechanism is through water buildup around the site. Thus, based on the determination that the general runoff is toward the east to Lake Michigan, the lake's size, and the relative level of the water in the lake with respect to the plant's elevation, it is concluded that no flooding of the lake from a combination of rain collection and runoff will ever endanger Kewaunee.

The licensee also stated [12] that the plant building roofs are designed to withstand a snow level of 40 pounds per square foot, the weight equivalent of 7.7 inches of water. Since the perimeter of each roof is equipped with a ledge that is approximately 3 inches higher than the roof surface, ponding would be limited to approximately 3 inches, which is well within design limits.

2.3.2.3 Significant Changes Since Issuance of the Operating License

The submittal does not identify any significant changes since the time of issuance of the plant OL.

2.3.2.4 Significant Findings and Plant-Unique Features

No significant findings are cited in the submittal. A summary of the walkdown procedures used by the licensee and the qualification of the team members performing the walkdown are not provided in the submittal.

2.3.2.5 Hazard Analysis

The external flooding hazard is deterministically screened out, therefore, no hazard analysis is performed.

2.3.3 Transportation and Nearby Facility Accidents

2.3.3.1 General Methodology

On Page 5-13 of the submittal it is stated that the methodology used for this analysis consists of first identifying the types and frequencies of hazardous material shipments. Next, an evaluation is made of the types of events involving hazardous material that could occur near the plant, and then their frequency of occurrence is estimated. Finally, the risk induced as the result of the occurrence of the postulated events is calculated. However, based on review of the examination as described in Section 5.3 of the submittal (Page 5-32), it seems that all hazards, other than air transportation accident hazards, were screened out based on deterministic calculations and data pertaining to the location of the plant with respect to the hazard under consideration. As such, the relevance of the methodology as stated in Section 5.0.3 of the submittal to the examinations reported in Section 5.3 of the submittal is not clear.

2.3.3.2 Plant-Specific Hazard Data and Licensing Basis

Since, according to the submittal, no large industrial plants are located nearby, the risk from nearby facilities is screened out. Ground transportation accidents via road and rail are identified as the only credible source of damage from off-site hazardous materials accidents. This hazard is screened out based on the significant quantity of chemicals that is needed to cause damage.

The danger from run-ground ships or barges collapsing the circulating water intake structure is identified as the only credible water transportation induced risk. It seems that this hazard is screened out based on the distance between alternate water supply lines and the maximum ship size that can cause damage to the intake structure. The explanation provided in the submittal is not clear, especially since no figures are provided.

Using the screening criteria presented in the NRC Standard Review Plan, the submittal concludes that the risk induced as a result of commercial and military flights is insignificant (in this case, less than 10^{-7} for the probability of exceedance of the radiological exposure guidelines set in 10 CFR Part 100). The examination of air transportation accidents is well-documented, and the results seem reasonable.

2.3.3.3 Significant Changes Since Issuance of the Operating License

The submittal does not identify any significant changes since the time of issuance of the plant OL.

2.3.3.4 Significant Findings and Plant-Unique Features

No significant findings are cited in the submittal. A summary of the walkdown procedures used by the licensee and the qualification of the team members performing the walkdown are not provided in the submittal.

2.3.3.5 Hazard Analysis

Air transportation accidents are the only hazard source explicitly screened out based on the hazard frequency. Based on the NRC SRP, this frequency was determined to be less than 10^{-7} per year. Other transportation events are deterministically screened out, and therefore, no hazard frequencies are reported for them.

2.3.4 Hazardous Materials

This analysis is basically a verification of the 1989 Updated Control Room Habitability Report, which was performed in response to NUREG/CR-0737 [24]. However, the analysis was further expanded to consider the effects of a release of hazardous materials on safety- related equipment or the local operation of the plant during emergencies. Note, these documents have not been examined as part of this review.

2.4 Generic Safety Issues (GSI-147, GSI-148, GSI-156 and GSI-172)

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

GSI-147 addresses the scenario of a 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

From the discussions provided in Section 4.1.3 of Reference [1], it can be inferred that fire-induced spurious actuation of components was considered. Since the submittal has followed the guidance provided in FIVE concerning control system interactions, all circuitry associated with remote shutdown is assumed to have been found to be electrically independent of the control room. Also, some information on this issue is provided in Section 4.8.1 of the submittal. It should be noted that the possibility of using the remote shutdown panel has not been modeled in the analysis.

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 [25] identifies possible reduction of manual fire-fighting effectiveness and causing misdirected suppression efforts as the central issue in GSI-148. As stated on page 54 of Reference [12], manual fire-fighting was not credited in the analysis. Thus, the issue of manual fire-fighting effectiveness is not addressed in this TER.

2.4.3 GSI-156, "Systematic Evaluation Program (SEP)"

Reference [25] provides the description of each SEP 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-156 may be found.

Settlement of Foundations and Buried Equipment

Description of the Issue [25]: The objective of this SEP issue is to assure that safety-related structures, systems and components are adequately protected against excessive settlement. The scope of this issue includes review of subsurface materials and foundations, in order to assess the potential static and seismically induced settlement of all safety-related structures and buried equipment. Excessive settlement or collapse of foundations could result in failures of structures, interconnecting piping, or control systems, such that the capability to safely shutdown the plant or mitigate the consequences of an accident could be comprised. This issue, applicable mainly to soil sites, involves two specific concerns:

- potential impact of static settlements of foundations and buried equipment where the soil might not have been properly prepared, and

- seismically induced settlement and potential soil liquefaction following a postulated seismic event.

Since static settlements are not believed to be a concern, the focus of this issue (when considering relevant information in IPEEEs) should be on seismically induced settlements and soil liquefaction. It is anticipated that full-scope seismic IPEEEs will address these concerns, following the guidance in EPRI NP-6041.

Kewaunee Nuclear Power Plant structures are founded on clay-sand soil deposits with an approximate depth to bedrock of 76 ft. Section 3.1.3.6 of the submittal provides a general discussion of soil properties and seismic soil failure analysis. The analysis of soil failures considers (1) soil liquefaction potential, and

(2) differential soil settlements/displacements under seismic conditions. In relation to the SPRA fragility calculations, Section 3.1.4.2 of the submittal provides additional discussion on the following topics: soil failure analysis and buried piping (submittal page 3-22); liquefaction (page 3-23); transient and permanent horizontal displacements and settlements (page 3-23); and buried piping from screenhouse and intake crib (pages 3-23 and 3-24). Additionally, Table 5-1 of the submittal provides some very brief remarks on the basis for screening out the following soil related hazards: coastal erosion (page 5-16), landslides (page 5-17), and soil shrink-swell consolidation (page 5-18).

Dam Integrity and Site Flooding

Description of the Issue [25]: The objective of this issue is to ensure the ability of a dam to prevent site flooding and to ensure a cooling water supply. The safety functions would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressures or erosion of soil materials, and providing sufficient freeboard and outlet capacity to prevent overtopping. Therefore, the focus is to assure that adequate safety margins are available under all loading conditions, and uncontrolled releases of retained water are prevented. The concern of site flooding resulting from non-seismic failure of an upstream dam (i.e., caused by high winds, flooding, and other events) is addressed as part of the SEP issue "site hydrology and ability to withstand floods." The concerns of site flooding resulting from the seismic failure of an upstream dam and loss of the ultimate heat sink caused by the seismically induced failure of a downstream dam should be addressed in the seismic portion of the IPEEE. The guidance for performing such evaluations is provided in Section 7 of EPRI NP-6041. As requested in NUREG-1407, the licensee's IPEEE submittal should provide specific information addressing this issue, if applicable to its plant. Information included for resolution of USI A-45 is also applicable to this concern.

The Kewaunee IPEEE submittal states, in Section 5.2.1, that there are no onsite or offsite dams associated with, or in the proximity of, the plant, and that dam failure and flooding from inland lakes and streams are not applicable to the plant site

Site Hydrology and Ability to Withstand Floods

Description of the Issue [25]: The objective of this issue is to identify the site hydrologic characteristics, in order to ensure the capability of safety-related structures to withstand flooding, to ensure adequate cooling water supply, and to ensure in-service inspection of water-control structures. This issue involves assessing the following:

- Hydrologic conditions - to assure that plant design reflects appropriate hydrologic conditions.
- Flooding potential and protection - to assure that the plant is adequately protected against floods.
- Ultimate heat sink - to assure an appropriate supply of cooling water during normal and emergency shutdown.

As requested in NUREG-1407, the licensee's IPEEE submittal should provide information addressing these concerns. The concern related to in-service inspection of water-control structures, a compliance issue, is not being covered in the IPEEE.

The Kewaunee IPEEE has included an evaluation of external floods (Section 5.2 of the submittal), including flooding on Lake Michigan and local flooding due to intense precipitation. The evaluation of flooding on Lake Michigan is presented in Section 5.2.3.A (pages 5-24 and 5-25) of the submittal, whereas the evaluation of flooding from intense precipitation is presented in Section 5.2.3.B (pages 5-25 and 5-26).

Industrial Hazards

Description of the Issue [25]: The objective of this issue is to ensure that the integrity of safety-related structures, systems, and components would not be jeopardized due to accident hazards from nearby facilities. Such hazards include: shock waves from nearby explosions, releases of hazardous gases or chemicals resulting in fires or explosions, aircraft impacts, and missiles resulting from nearby explosions. As requested in NUREG-1407, the licensee's IPEEE submittal should provide information addressing this issue.

The Kewaunee IPEEE submittal (Section 5.3) includes the following information of relevance to this issue: Section 5.3.1.A of the submittal discusses the potential for accidents at nearby industrial facilities; Section 5.3.1.B discusses potential ground transportation accidents; Section 5.3.1.C discusses potential water transportation accidents (due to ships or barges running aground); Sections 5.3.1.D and 5.3.3 discuss potential air transportation accidents; and Section 5.4 discusses potential onsite and offsite hazardous material accidents.

Tornado Missiles

Description of the Issue [25]: The objective of this issue is to assure that plants constructed prior to 1972 (SEP plants) are adequately protected against tornadoes. Safety-related structures, systems, and components need to be able to withstand the impact of an appropriate postulated spectrum of tornado-generated missiles. As requested in NUREG-1407, the licensee's IPEEE submittal should provide information addressing this issue.

The Kewaunee IPEEE has involved an evaluation of tornadoes, as documented in Section 5.1.3.B of the submittal. Although tornado-induced missiles are mentioned briefly in this evaluation (page 5-22), no analysis of tornado-induced missiles was performed.

Severe Weather Effects on Structures

Description of the Issue [25]: The objective of this issue is to assure that safety-related structures, systems, and components are designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include: straight wind loads, tornadoes, snow and ice loads, and other phenomena judged to be significant for a particular site. As requested in NUREG-1407, the licensee's IPEEE submittal should provide information specifically addressing high winds and floods. Other severe weather conditions (i.e., snow and ice loads) were determined to have insignificant effects on structures (see Chapter 2 of NUREG-1407).

The Kewaunee IPEEE has included evaluations of high winds (straight wind loads and tornadoes) and external floods. Section 5.1 of the submittal discusses severe winds and tornadoes, and Section 5.2 of the submittal discusses external floods. In addition, Table 5-1 of the submittal provides brief justifications for screening out some severe weather effects, such as drought, fog, hail, lightning, snow, ice, low winter temperatures, etc.

Design Codes, Criteria, and Load Combinations

Description of the Issue [25]: The objective of this issue is to assure that structures important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with their safety function. All structures, classified as Seismic Category I, are required to withstand the appropriate design conditions without impairment of structural integrity or the performance of required safety functions. Due to the evolutionary nature of design codes and standards, operating plants may have been designed to codes and criteria which differ from those currently used for evaluating new plants. Therefore, the focus of this issue is to assure that plant Category I structures will withstand the appropriate design conditions (i.e., against seismic, high winds, and floods) without impairment of structural integrity or the performance of required safety function. As part of the IPEEE, licensees are expected to perform analyses to identify potential severe accident vulnerabilities associated with external events (i.e., assess the seismic capacities of their plants either by performing seismic PRAs or SMAs).

The Kewaunee IPEEE has included an evaluation of potential severe accident vulnerabilities associated with external events. The submittal does not systematically identify codes, criteria, and load combinations used in design. Section 3.1.4.2 of the submittal provides some brief information on the seismic Category classification and seismic design of building structures (page 3-22), and Section 3.1.2 provides some general information related to seismic design loads. Wind design speeds are cited in Section 5.1.3.

Seismic Design of Structures, Systems, and Components

Description of the Issue [25]: The objective of this SEP issue is to review and evaluate the original seismic design of safety-related structures, systems, and components, to ensure the capability of the plant to withstand the effects of a Safe Shutdown Earthquake (SSE).

The Kewaunee IPEEE is based on a seismic PRA, which has evaluated failure probabilities of the plant and plant structures, systems, and components, at various ground motion levels. The related probabilistic analyses are documented in Sections 3.1.2 to 3.1.5 of the submittal.

Shutdown Systems and Electrical Instrumentation and Control Features

Description of the Issue [25]: The issue on shutdown systems is to address the capacity of plants to ensure reliable shutdown using safety-grade equipment. The issue on electrical instrumentation and control is to assess the functional capabilities of electrical instrumentation and control features of systems required for safe shutdown, including support systems. These systems should be designed, fabricated, installed, and tested to quality standards, and remain functional following external events. In IPEEEs, licensees were requested to address USI A-45, "Shutdown Decay Heat Removal (DHR) Requirements," and to identify potential vulnerabilities associated with DHR systems following the occurrence of external events. The resolution of USI A-45 should address these two issues.

The licensee provides a detailed discussion of its treatment for resolution of USI A-45 for external events (i.e., seismic, fire, and other events) in Section 3.2 (pages 3-48 to 3-61) of the Kewaunee IPEEE submittal. Section 4.9 of the submittal mentions the fire IPEEE consideration of the USI A-45 issue, citing the discussion in Section 3.2. Sections 2.1.15 and 2.2.13 of this TER summarize review findings related to USI A-45, respectively, for seismic events and fire events.

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

Reference [25] 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 [25]: 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.

Information related to operator recovery actions following seismic events is provided on page 3-8, page 3-34, and pages 3-39 to 3-43 of the submittal. The submittal addresses operator recovery actions for fire events in Sections 4.5 and 4.6. Additionally, the submittal's discussion of the USI A-45 issue, in Section 3.2 (pages 3-48 to 3-61), provides information on operator recovery actions.

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

Description of the Issue [25]: 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 Kewaunee 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 [25]: Fire can damage one train of equipment in one fire zone, while a redundant train could potentially be damaged in one of the 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 Kewaunee 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. Some brief information specifically related to GI-57 is provided in Section 4.9 (page 4-50) of the submittal.

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

Description of the Issue [25]: 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.

Some information pertaining to suppression-induced damage to equipment, as well as seismically induced inadvertent actuation of fire suppression systems, can be found, respectively, in Sections 4.8.3 and 4.8.5 of the IPEEE submittal.

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

Description of the Issue [25]: 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 Kewaunee IPEEE submittal discusses external floods in Section 5.2, and has some discussion of non-seismic actuations of fire suppression systems in Section 4.8.3, and of seismically induced inadvertent actuation of fire suppression systems in Section 4.8.5. Other than the information pertaining to seismically induced inadvertent actuation, the submittal does not provide systematic discussion of seismically induced floods.

Seismically Induced Spatial and Functional Interactions

Description of the Issue [25]: 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 Kewaunee 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 (page 3-6), 3.1.2.3, 3.1.3, and 3.1.4 of the submittal.

Seismically Induced Fires

Description of the Issue [25]: 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.

The Kewaunee IPEEE submittal provides a brief discussion of seismically induced fires in Section 4.8.5.1.

Seismically Induced Fire Suppression System Actuation

Description of the Issue [25]: 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.

Some information pertaining to seismically induced inadvertent actuation of fire suppression systems can be found in Sections 4.8.5.2 and 4.8.5.3 of the IPEEE submittal.

Seismically Induced Flooding

Description of the Issue [25]: 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 submittal provides no distinct discussion of seismically induced flooding, other than information related to seismically induced inadvertent actuation of fire protection systems (as addressed in Sections 4.8.5.2 and 4.8.5.3). Page 3-24 of the submittal indicates that small bore pipe were reviewed during plant walkdowns. The submittal is not clear as to what extent non-seismically designed piping or tanks were considered in the seismic analysis.

Seismically Induced Relay Chatter

Description of the Issue [25]: 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.

As noted in Sections 3.1.4.2 (page 3-21) and 3.1.6.3.C (page 3-46) of the Kewaunee IPEEE submittal, a relay chatter analysis for Kewaunee was performed as part of USI A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants." The extent, if any, to which relay chatter impacts were modeled in the seismic PRA is not discussed in the submittal.

Evaluation of Earthquake Magnitudes Greater than the Safe Shutdown Earthquake

Description of the Issue [25]: 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 Kewaunee IPEEE has included a seismic PRA, as documented in Section 3 of the submittal. As noted in this TER, however, the evaluated plant high-confidence of low-probability of failure capacity spectrum does not exceed even the plant's design (SSE) spectrum over some important frequency ranges.

Effects of Hydrogen Line Ruptures

Description of the Issue [25]: 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.

The Kewaunee IPEEE submittal provides no discussion concerning the potential and effects of hydrogen line and tank ruptures.

3. OVERALL EVALUATION, CONCLUSIONS AND RECOMMENDATIONS

3.1 Seismic

For the seismic IPEEE of Kewaunee, the licensee's SPRA approach, especially the manner of usage of the surrogate element, requires special consideration because: (a) it appears to satisfy NUREG-1407 guidelines for an NRC-type SMA, and yet (b) it does not reveal the true dominant risk contributors (because these have capacities in excess of the surrogate element capacity). No deficiencies in scope of evaluation are apparent in the submittal; i.e., the seismic IPEEE of Kewaunee Nuclear Power Plant addresses the major elements specified in NUREG-1407 as recommended items that should be considered for evaluation of a focused-scope plant. The submittal itself gives a clear description of the seismic evaluation, and the documentation is considered to be well-written. The identification and implementation of safety enhancements, as a result of the plant walkdowns, has produced some meaningful insights in response to the objectives of GL 88-20, for a focused-scope plant. Seismic fragilities and HCLPF capacities have provided valuable information on the capability of plant components.

Even though the Kewaunee seismic IPEEE is judged to be essentially complete with respect to the guidelines and objectives of NUREG-1407 and of GL 88-20, there are some problems pertaining to implications of the licensee's SPRA evaluation of Kewaunee that need to be mentioned in this TER. First, as noted above, the surrogate element has been found to be a dominant contributor to seismic CDF. This finding, however, is largely artificial and substantially misleading for the case of Kewaunee Nuclear Power Plant. Components that are screened out in the SPRA are represented by the surrogate element; yet such screened-out components are expected to be more seismically resistant than components which can not be screened out. In the Kewaunee SPRA, all components that have not been screened out have been determined as having a capacity about equal to, or greater than, the surrogate element capacity. Consequently, the actual dominant CDF contributors are being masked by the surrogate element. In other words, the surrogate element capacity acts as a limiting threshold, above which assessment of dominant risk contributors is meaningless. The potential for the surrogate element to mask the true dominant CDF contributors is minimized when several components are screened in and found to have capacities significantly lower than the surrogate element capacity. In such cases, the surrogate element capacity threshold is still high enough for useful insights to be obtained pertaining to true risk contributors. For Kewaunee, though, this is not the situation. However, because the surrogate element capacity also represents a screening threshold, above which components can be said not to be outliers, any dominant CDF contributors that might exist above this level are of questionable importance. That such contributors are of questionable importance (at least for purposes of IPEEE resolution) can be best seen by viewing the SPRA as a (NRC-type) seismic margins assessment. An SMA is an appropriate option for IPEEE implementation for Kewaunee. With respect to vulnerability assessment based on an SMA, only components found to have capacities less than the review level earthquake would be designated as outliers requiring further resolution/treatment. Components having capacity in excess of the RLE are generally viewed as being sufficiently robust, and hence, would not need to be evaluated in additional detail. Hence, even though the Kewaunee SPRA-related insights with respect to dominant contributors may be minimal or unrealistic, the Kewaunee seismic IPEEE can still be viewed as producing substantially valid SMA-type findings.

A second significant observation pertains to the meaning of reported HCLPF capacities. NUREG-1407 requests, for an SMA, that the median NUREG/CR-0098 spectral shape, anchored to the RLE peak ground acceleration of 0.3g, be used for evaluating HCLPF capacities. For an SPRA, however, it is considered acceptable to use a site-specific spectral shape -- for instance, the LLNL 10,000-yr median UHS shape -- to characterize motion input. This latter spectral shape was used for assessing the Kewaunee HCLPF capacities. Strictly speaking, this usage somewhat undermines the preceding conclusion that the Kewaunee seismic IPEEE can be viewed as producing valid SMA-type findings. An SMA for seismic IPEEE implementation should, in accordance with NUREG-1407, evaluate HCLPF capacities with respect to the more-conservative NUREG/CR-0098 shape. The Kewaunee seismic IPEEE reports a plant-level HCLPF capacity of 0.26g (seismic failures only) with respect to the LLNL UHS shape. However, being derived fundamentally from the EPRI SMA spectral acceleration screening limits, the surrogate element capacity is, in fact, complete with respect to the ordinates of the NUREG/CR-0098 spectrum. Thus, if the surrogate element indeed dominates the plant-level HCLPF capacity, the capacity may actually be reported with respect to the NUREG/CR-0098 shape. To the extent that other, screened-in components (whose capacities are, in fact, evaluated with respect to the LLNL UHS shape) also influence the plant-level HCLPF to some extent, the capacity may not be complete with respect to ordinates defined by the NUREG/CR-0098 spectrum.

A third observation relates to the findings from sensitivity studies, conducted as part of the Kewaunee seismic IPEEE, with respect to the importance of human actions. These studies have been used to suggest that human errors do not have a controlling influence on CDF nor on containment performance. However, the seismic IPEEE submittal notes elsewhere that an operator failure is a dominant CDF contributor. In fact, the significant influence of human error on the potential for both core damage and failure of containment safeguards, can be clearly seen in the reported fragility curves. It is judged that simultaneous variation of parameters in sensitivity studies could have revealed human errors as having an important influence on CDF. However, based on the present technical review, the human error fragilities developed in the IPEEE are themselves not considered to be very realistic (for example, they do not depend on where and when the required human actions must be performed). Clearly, though, regardless of the development of more realistic human error rates, meaningful findings pertaining to the importance of human errors are likely to be obscured by the study's use of the surrogate element.

As judged from the present submittal-only review, the following items are viewed as strengths of the seismic IPEEE submittal for Kewaunee:

<u>Strengths</u>

1. The level of analysis (i.e., seismic PRA) employed for the overall seismic IPEEE process. (However, it is most practical to view the study as, first, an NRC-type SMA, and second, as an SPRA. This is because SMA-type conclusions regarding component screening and component capacities are considered robust, but not the SPRA-type conclusions regarding dominant contributors and dominant accident sequences.)

2. The clear and comprehensive explanatory detail provided in the seismic IPEEE submittal and in additional documentation provided by the licensee.

3. The effort put forth in plant seismic walkdowns, and the resulting insights achieved concerning non-design-related plant deficiencies.

4. The overall plant modeling process and some useful sensitivity analyses. In particular, the format of comparisons for system-level fragilities, sequence-level fragilities, and plant-level fragilities is extremely useful in understanding the IPEEE results.

5. The degree of licensee participation in the seismic IPEEE process and licensee's intent to make the IPEEE a living study.

Even though the Kewaunee seismic IPEEE is judged to be essentially complete with respect to the guidelines and objectives of NUREG-1407 and of GL 88-20, as noted above, there are some problems pertaining to implications of the licensee's SPRA evaluation of Kewaunee. The most significant observations/conclusions that pertain to limitations of the seismic IPEEE insights, are noted as follows:

Weaknesses

1. The manner of usage of the surrogate element for the Kewaunee IPEEE (i.e., where there are few screened-in components having seismic capacities less than the surrogate element capacity) does not produce valid PRA insights/findings. As a consequence, a meaningful set of dominant contributors has not been found.

2. Component and plant-level HCLPF capacities are reported with respect to a UHS shape, as opposed to a NUREG/CR-0098 spectral shape (the spectral shape recommended in NUREG-1407 for reporting HCLPF capacities). The current plant HCLPF spectrum (with 0.23g or 0.26g PGA), therefore, does not exceed even the plant design spectrum over some important frequency ranges.

3. Fragilities characterizing human error rates are not realistic, and have not been based on a fundamental consideration of where and when the required human actions should be performed.

4. The study has not proposed improvements to procedures which would reduce the potential for the following operator errors:

   a. Operator failure to shift AFW pumps from the CST to service water.
   b. Operator failure to open manual valve ICS-7A or ICS-7B after testing.
   c. Operator failure to initiate Internal Containment Spray (ICS) recirculation.

5. Safety enhancements to the RHR heat exchangers, and an evaluation of resulting impacts on seismic CDF, have not been considered.

3.2 Fire

The licensee has expended a considerable effort in the preparation of the fire hazard part of the IPEEE. The IPEEE report complies with the conditions set forth in Reference [3]. The licensee has employed a proper methodology and database for conducting the fire analysis. A combination of the FIVE and Level-1 fire PRA methodologies have been used, including a screening procedure.

The following are strengths of the submittal:

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

2. The final core damage frequencies can be traced back to the initial assumptions and frequencies. The reviewers were able to trace some of the calculations through the analysis.

3. State-of-the-art methodology and proper data have been used.

4. The study includes a qualitative analysis of the uncertainties. The study event tree/fault tree methodology is sound, and the selection of initiating events makes sense. The study's logic in the development of initiating event frequencies, and in combining fire frequencies with suppression/detection, is sound. The study's screening analysis appears sound.

The following is a list of weak areas of the analysis that are deemed to be sufficiently important to be brought to the licensee's attention:

1. The study does not justify the selection of locations for COMPBRN runs; nor does it provide the reasons for ignoring many other locations within a room. One possible justification would be to perform sensitivity runs for other locations, and show that the results are not worse than those selected for fire propagation analysis.

2. The initiating event frequencies are determined by rationing the generic zone/area frequency with the fire influence area from COMPBRN. Thus, the total fire frequency may be sensitive to the number of COMPBRN locations chosen. Again, this points toward the need for strong justification of selection and elimination of fire locations.

3. Lack of use of plant-specific data could lead to non-conservative results. Each diesel generator room (there are two of them) has a calculated fire initiation frequency of 0.01 per year. However, the fire data records show that Kewaunee experienced a significant fire in a diesel generator room in 1977, owing to carbon buildup in the exhaust path. The plant went operational in 1974. Thus, there has been 1 fire in 40 diesel generator room-yrs. Perhaps a Bayesian update would help to reconcile the generic with the plant-specific numbers, or perhaps that fire is no longer applicable because of changes to the diesel generator or changes in its testing protocol.

   Kewaunee also experienced a fire in 1975 caused by a bus fault in the main auxiliary transformer bay that caused a reactor trip. This is usually in or adjacent to the 4160 V switchgear room. Generic data for fire in that room is given as 9×10^{-4}/yr. Again, there needs to be reconciliation between the plant-specific experience and the low generic number.

4. The study claims that damage owing to inadvertent suppression system actuation is not a problem, since suppression systems in safety critical areas are not charged. The discussion of GI-57 on Page 4-50 of Reference [1] makes the statement, "since automatic fire protection systems are minimally present in areas where safety-critical equipment resides, there is minimal impact if fire protection systems were to actuate." This statement is incorrect. Automatic fire protection equipment is present in several safety-critical areas, including the fire risk significant areas. For example, automatic suppression equipment is present in the DG rooms and the cable spreading

room. The relay room has manually actuated total CO_2 flooding, and the cable spreading room has wet pipe sprinklers. Furthermore, in other areas preaction system flow control valves may fail from a seismic event.

5. Most of the auxiliary building (AX-23A and AX-23B) is screened out because it is a large open area. This is not sufficient justification, since direct radiant energy from a fire near safety-related equipment could cause damage.

6. The assumption that all automatic fire suppression systems are perfectly designed is not reasonable. A plant walkdown must verify that not only is the size of the system correct, but that the placement and elevation with respect to fire sources and targets is adequate. Also, the unavailability of the suppression system should be properly accounted for in all PRA-style fire analyses.

7. The assumption that fire barriers remain intact for their rated time may not be accurate, due to an open door or vents/ducts. The unreliability of active fire barriers (e.g., a normally open, self closing fire door or normally open fire dampers) should be considered in defining adequacy of fire zone separation. Another example can be brought from a large pump area that contains a considerable amount of oil. An oil leak may spray ignited oil over a distance and cross over a normally open fire barrier.

   This assumption led the study team to not adequately investigate cross-zone fire propagation.

8. The study's argument for screening out the diesel generator day tank room (TU-91) omits the potential for transient combustibles to start a large fire.

9 On the top of Page 4-45, the submittal claims that damage due to fire effects or suppression activities was assumed to fail affected components. However, the study did not identify any such components.

10. The assumption in the control room fire analysis regarding blown fuses protecting control circuits from the effects of a control panel fire may not be valid. The action of a switch may be simulated if two wires contact one another because of fire damage to their insulation. The fuse for this circuit would not blow under such conditions. However, it must be noted that if the operators remove the fuses because of procedural requirements, the effect may ultimately be the same.

3.3 <u>HFO Events</u>

The submittal generally followed the submittal guidelines and basic steps of analyzing and reporting potential accident scenarios. Two specific weaknesses, however, have been identified by this review. These are summarized as follows:

1. The frequency of a tornado striking in the vicinity of the plant is estimated to be 4.86×10^{-4}/yr (Page 5-22 of Reference [1]), which is above the screening level. However, the risk induced from tornadoes is screened out on the basis that the frequency of occurrence of tornadoes with wind speed greater than the plant's design basis wind speed is negligible (Page 5-23 of Reference [1]). No discussion of the potential hazard posed by tornado generated missiles is provided in the

submittal. However, the tornado missile analysis documented in the plant USAR concludes that missile impact load is unlikely to cause damage to Class I structures according to the applied design criteria. No discussion is provided as to whether or not there are non-Class I structures of importance to the IPE conclusions which may not be protected against tornado missiles. The licensee has also indicated that during a safety system functional inspection for the emergency diesel generators, the design of the vents on the underground diesel oil storage has been identified as an open item, and scheduled for resolution during 1996.

2. The submittal states that Kewaunee has facilities that were designed and built prior to the current NRC criteria (Page 5-11). However, no specific facility is identified.

4. IPEEE INSIGHTS, IMPROVEMENTS AND COMMITMENTS

4.1 Seismic

Calculations of seismic capacities for screened-in items have revealed one component HCLPF estimate to be as low as 0.29g, i.e., for the RHR heat exchangers. The overall plant HCLPF capacity has been reported to be equal to 0.23g, accounting for non-seismic failures and human errors, and 0.26g when the non-seismic failures and human errors are ignored. The HCLPF capacity of containment to prevent a large early release was assessed at 0.30g. (These HCLPF assessments have been reported with respect to the LLNL median 10,000-yr UHS shape; this spectral shape is significantly different from the NUREG/CR-0098 median, 5%-damped spectral shape which is recommended in NUREG-1407 as the basis for reporting HCLPF capacities in a seismic margin assessment.) CDF values of 1.10×10^{-5}/ry and 1.15×10^{-5}/ry have been reported, respectively, for the 1989 EPRI hazard results and the 1993 LLNL hazard results. (These CDF values are for truncated seismic hazard curves. A CDF value of 1.32×10^{-5}/ry was reported for the untruncated 1993 LLNL hazard results.) The contaiument seismic failure frequency has been estimated at 6.24×10^{-6}/ry.

The submittal states that there are no seismic vulnerabilities at Kewaunee Nuclear Power Plant. No major plant changes were deemed necessary by the licensee based on the results of the Kewaunee IPEEE. The seismic IPEEE of Kewaunee did, however, identify several open issues requiring resolution. The open issues are identified in Table 3-4 of the Kewaunee IPEEE submittal -- which is repeated here as Table 4.1 -- together with their disposition status. Sixteen different outliers/issues are noted in this table. Some equipment enhancements, one procedural implementation, an administrative control, and several housekeeping improvements resulted from the study. The equipment enhancements included: installing missing fasteners on DG excitation and control cabinets, upgrading anchorage of station service transformers, bolting together relay racks, and implementing design changes for equipment anchorages and mercoid switches. The various plant enhancements have been either implemented or scheduled for implementation.

According to the submittal, the dominant basic events/component failures that contribute to seismic risk are:

1. Loss of offsite power

2. Surrogate component in the following systems: Containment or Steam Generator failure, Reactor Vessel or Building failures, Service Water System, DC Power System, and AC Power System

3. Operator Error - failure to shift Auxiliary Feedwater (AFW) pumps from the Condensate Storage Tank (CST) to service water

Based on the SPRA results (including sensitivity evaluations), the following conclusions are arrived at by the licensee:

a. There does not exist a single failure mode for Kewaunee that dominates the seismic core damage frequency.

b. Failure of a surrogate component, which is a conservative measure of the capacity of components that were screened out, is for many systems the important mode of failure. Since the surrogate component does not specifically model the failure of a particular component, this observation is a further reinforcement of the conclusion that there does not exist any component specific failure modes that dominate the seismic CDF.

c. Operator actions are not a major contributor to the seismic CDF or plant capacity.

d. Loss of offsite power is an important contributor to the seismic risk.

e. As a group, random failures and operator actions are an important part of the seismic CDF. In a relative sense, variation in the random failure probabilities produced the largest change in the seismic CDF (a range corresponding to a factor of 2.5).

f. For seismic containment performance, the results of the SPRA evaluations indicate that the containment, as well as the systems designed to ensure containment integrity, are seismically sound, and no vulnerabilities could be identified.

The finding that the surrogate element is a dominant risk contributor is considered to not be a meaningful result, as screened-out components are not expected to control plant seismic failure. In addition, the findings that an operator error was found to be a dominant basic event (Item 3 above), and yet operator actions have been deemed not to be major contributors to seismic CDF (Item c above), are inconsistent. Operator actions can be seen to have a major effect on Kewaunee plant-level fragility curves pertaining both to core damage and to failure of containment safeguard systems; however, the operator fragility curves used in the IPEEE are not considered to be highly meaningful.

A detailed discussion, based on PRA methods and findings, is provided in the IPEEE submittal pertaining to USI A-45 resolution for external events. No plant vulnerabilities were identified as a result of the USI A-45 evaluation.

GI-131 is not, strictly speaking, applicable to Kewaunee, because the flux mapping cart is not movable. However, the lateral resistance of the mapping system was evaluated to be seismically adequate. In addition, an administrative control was implemented to insure proper restraint of a chain hoist, in order to eliminate a potential interaction hazard with the ten-path assembly of the flux mapping system.

A significant effort in coordination of USI A-46 and the seismic IPEEE has taken place for the seismic evaluation of Kewaunee. USI A-46 is resolved separately from the seismic IPEEE. The submittal notes that resolution of USI A-17 and USI A-40 will be addressed in the USI A-46 submittal.

4.2 Fire

The licensee states that "in general, no significant fire concerns were discovered." The total core damage frequency is estimated to be 1.81×10^{-4} per reactor year. The dominant core damage frequencies are associated with fires in the two auxiliary feedwater pump rooms, the cable spreading room, one of the diesel rooms and the control room. These fire events can simultaneously render several diverse or redundant components inoperable. The entire exercise 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 will be necessary to protect the core from any adverse effects.

4.3 HFO Events

The IPEEE's overall conclusion regarding this category of external events is that any potential core damage scenario has an extremely low frequency in comparison with the frequency of core damage from other initiators. As a result, no safety enhancements have been identified. However, the licensee has indicated that during a safety system functional inspection for the emergency diesel generators, the design of the vents on the underground diesel oil storage has been identified as an open item and scheduled for resolution during 1996.

Table 4.1 Equipment Outliers/IPEEE Walkdown Results (Submittal Table 3-4)

Table 3-4 EQUIPMENT OUTLIERS IPEEE WALKDOWN RESULTS		
EQUIPMENT DESCRIPTION	**FINDING**	**RESOLUTION**
Motor Control Centers MCC52F & MCC52FEXT.	Adjacent MCC's not bolted together, which may pose an interaction hazard based on relay chatter concerns.	Cabinet displacements during a design basis seismic event were determined. The evaluation concluded that the cabinets will not impact.
Diesel Generator Excitation & Control Cabinets DR101 & DR111.	Several fasteners on cast-in-place anchors were found missing. An overhead emergency light posed an interaction hazard to DR101.	Missing fasteners were replaced during the 1992 refueling outage and restraint installed on emergency light during 1993 outage.
Station Service Transformers 51, 52, 61 & 62.	Transformer cabinets were found anchored to the floor with friction clips, which are considered undesirable according to A-46 walkdown guidelines.	A design change was initiated to have the transformer cabinet bases welded to embedded floor channels. Transformers 51 and 52 were modified during the 1994 refueling outage. Transformers 61 and 62 are scheduled for modification during a later refueling outage.
Relay Racks RR186 & RR187.	The relay racks are not bolted to adjacent panels, which may pose an interaction hazard based on relay chatter concerns.	An engineering support request was initiated to have the racks bolted together to eliminate the concern. A schedule for completion has not been determined.
Reactor Trip Breaker Cabinet RD106.	Several anchor bolts which connect cabinet to embedded channel were found missing.	A design change was initiated to have the cabinet sections welded to the embedded channel in lieu of installing bolts. Work was completed during the 1993 refueling outage.

Table 3-4
EQUIPMENT OUTLIERS
IPEEE WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
SI Pump B Suction Isolation Valve SI5B.	One leg of a Unistrut frame is within 1/2" of the valve motor, which may present an interaction hazard.	IE Bulletin 79-14 pipe stress evaluation determined that displacement of the pipe and valve is approximately 1/8". Issue considered resolved.
Main Steam Header 1A Controlled Relief Valve SD3A.	Valve actuator and yoke are independently braced.	An analysis was performed to qualify support configuration as-is.
SI Pump Makeup Valve SI101B to Accumulator.	Actuator and yoke are independently braced.	An analysis was performed to qualify support configuration as-is.
Aux Feedwater Pump Lube Oil Pressure Switches 16016, 16019 & 16085.	All three switches identified as Mercoids, which are considered outliers for the A-46 program.	A design modification was previously initiated to have the switches replaced for other reasons. Modification completed in 1993.
Flux Mapping Transfer Cart (GI-131 Issue).	Two concerns identified; (1) lateral restraints for the 10-path assembly frame were never installed, and (2) chain hoist on overhead rail identified as a possible interaction hazard to 10-path assembly.	It was determined by analysis that lateral restraints are not required to support 10-path assembly under seismic loads. Administrative controls were implemented to restrain hoist at the fixed end of crane rail and required to be functional when not in use.
Overhead Fluorescent Lights.	Generic problem throughout safety-related areas of the plant. S-hooks on the chains supporting the lights are not closed, presenting a possible interaction hazard to equipment below.	A plant walkdown was conducted during the 1994 refueling outage to pinch the S-hooks closed.

Table 4.1 Equipment Outliers/IPEEE Walkdown Results (Submittal Table 3-4) (Continued)

Table 3-4
EQUIPMENT OUTLIERS
IPEEE WALKDOWN RESULTS

EQUIPMENT DESCRIPTION	FINDING	RESOLUTION
Emergency Lights.	Some of the lights were found to not have seismic restraints installed, presenting a possible interaction hazard to equipment below.	Lights and battery units strapped to supports as required during 1994 refueling outage.
480V Switchgear Bus 62.	An empty spare breaker cabinet was used for parts storage, presenting a possible interaction hazard on the basis of relay chatter concerns.	Maintenance department notified of problem. Spare parts were removed and all other spare breaker cabinets were inspected for similar problems during 1993 refueling outage.
Control Room Ceiling.	Aluminum ceiling diffuser panels were considered as a possible hazard to operators if the diffusers were to dislodge from T-bar supports.	An engineering support request was initiated to have the diffuser panels tie-wrapped to the T-bar supports. A schedule for completion has not been determined.
Control Room Vertical Panel C.	Rear doors on panel could not be latched shut due to interference with cables which extend from rear of cabinet. Unlatched doors present possible interaction hazard on basis of relay chatter.	It was determined that the cables were temporarily in place to support radiation monitoring modifications. Doors could not impact with cabinet because of cable interference. Operations department agreed to latch doors shut following completion of work during 1994 refueling outage.
All Equipment	Possible interaction hazards due to loose or unrestrained portable equipment.	Plant procedure GNP 1.31.1 drafted to provide guidelines for control of portable equipment. Full implementation occurred June 1, 1994.

5. IPEEE DATA SUMMARY AND ENTRY SHEETS

Completed data entry sheets for the Kewaunee IPEEE are provided in Tables 5.1 to 5.6. These tables have been completed in accordance with the descriptions in Reference [11]. Table 5.1 lists the overall external events results. Table 5.2 summarizes the important seismic PRA fragility values. Tables 5.3 and 5.4 provide the PWR Accident Sequence Overview and Detailed tables for seismic events, respectively. Tables 5.5 and 5.6 provide the PWR Accident Sequence Overview and Detailed tables for fire events, respectively. Accident sequence tables are not provided for HFO events, since no PRA analyses were performed for this class of events.

Table 5.1
External Events Results

Plant Name: Kewaunee

Event	Screening	CDF	Plant HCLPF(g)	Notes
External Fire	O			
External Flooding	O			
Extreme Winds	O			
Internal Fire	S	1.81E-04/ry		
Nearby Facility Accidents	O			
Seismic Activity	S	1.32E-05/ry	0.23	CDF=1.32E-05 (1993 LLNL Untruncated) CDF=1.15E-05 (1993 LLNL Truncated) CDF=1.10E-05 (EPRI) HCLPF=0.23g (with non-seismic failures) HCLPF=0.26g (without non-seismic failures) HCLPF=0.30g (containment failure with non-seismic failures)
Transportation Accidents	O			
Others	O			Hazardous Materials

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

Table 5.2
PRA Seismic Fragility

Plant Name: Kewaunee SSE: Horizontal 0.12 (g) Vertical ____ (g)

Hazard parameter: PGA (PGA, Spectral Velocity)

Hazard Assessment: 1993 LLNL EPRI (LLNL, EPRI, Site Specific)

Spectral Shape: 1989 LLNL 10,000 yr UHS (10,000 year LLNL median UHS, site specific or other)

Cutoff "g": 1.0g (1993 LLNL); Infinite (1993 LLNL); 1.6g (EPRI)

List components and equipments with lowest seismic capacities (less than 10) which contribute to system failure:

Component	Median Capacity (g)	β_R	β_U	β_C	HCLPF (g)	Seismic Sequence Description	Seismic Success Path Description
Surrogate Element	0.64			0.30			
RHR Heat Exchangers	0.63			0.46			
Circulating Water Intake and Discharge Piping	1.05			0.46			
Motor Control Center MCC 1-62J	1.08			0.46			
Distribution Cabinets	1.10			0.46			
Relay Rack - Fuse Panel	1.11			0.46			
Boraic Acid Tank	1.16			0.46			
Turbine Building Fan Coil	1.42			0.46			
Residual Heat Pump Fan Coil	1.43			0.46			
Offsite Power[1]	0.35			0.55			

[1] Not a plant component.

Table 5.3
PWR Accident Sequence Overview Table

Plant Name: Kewuanee — For Seismic PRA Only — 1 Sheet of 1

#	Sequence	PDS	CDF	Init Event	Loss Support	Failed Functions	Attributes
1	SCSF	LFFFF	3.60E-6/ry	A		RCS-INT, LPI, CIF	BYPASS
2	SSWS	Note 1	1.99E-6/ry	T-LOOP	ESW	SSMU, RCS-DEP	SBO, TIL
3	SRVB	LFFFF	1.74E-6/ry	Note 2	Note 2	Note 2	Note 2
4	SLSP01	Note 1	1.35E-6/ry	T-LOOP		SSMU, RCS-DEP	TIL, HUM
5	SACP	Note 1	1.26E-6/ry	T-LOOP	EAC	SSMU, RCS-DEP	SBO, TIL
6	SDCP	Note 1	3.48E-7/ry	T-LOOP	EAC	SSMU, RCS-DEP	SBO, TIL

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

Loss Supports: At most two of the following: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC3, AUXC4, CCW, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, OA3, OA4, SA, STM, SW2, SW3, SW4, VAC (Field may be blank).

Failed Functions: At most three of the following: SINT, SDEP, SSMU, 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, TIL, IND-SGTR, SBO, OR HUM (Field may be blank)

Notes:
1. Applicable PDS is not evident from information provided in the submittal.
2. Situation specific (since SRVB encompasses a number of different initiators).

Table 5.4
PWR Accident Sequence Detailed Table

Plant Name: Kewaunee

For Seismic PRA Only

#	SEQUENCE	RX		PRIMARY INTEGRITY					PRIMARY INVENTORY-INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT										NOTES
		RPS	BI	PPORV	PSRV	PAD1	PAD2	RCPS	CHP1	HPI	LPI	ACC	AI1	AI2	CHPR	HPR	LPR	AR1	AR2	SGS	SGA	TT	MSIV	TB	SG	MFW	NISP	AFW	AM1	AM2	AM3	CS1	CS2	FC1	FC2	IC	CI1	CI2	IGN	RF	HUM	
1	SCSF																																									Gross containment/RPV failure
2	SSWS																																									Unrecoverable SBO
3	SRVB																																									Includes four different building failures, all leading directly to CD
4	SLSP01			X																								X													X	
5	SACP																																									Unrecoverable SBO
6	SDCP																																									Unrecoverable SBO

Table 5.5
PWR Accident Sequence Overview Table

Plant Name: Kewuance For Fire PRA Only 1 Sheet of 1

#	Sequence	PDS	CDF	Init Event	Loss Support	Failed Functions	Attributes
1	Fi6-AF3-OB5	Early Core Damage	4.21E-5/ry	T-RX	Offsite Power, Diesel Generator, Service Water	RCS-DEP, HPI, SSMU	TIL, HUM, SBO
2	Fi4-CHP-CCL	Late Core Damage	3.73E-5/ry	S2	Diesel Generator, CCW	HPI	TIL, HUM
3	Fi7-AF3	Early Core Damage	2.91E-5/ry	T-RX	Service Water, Diesel Generator	SSMU, HPI	TIL, HUM
4	Fi2-AF3	Early Core Damage	2.42E-5/ry	T-RX	Service Water, AFW	SSMU, HPI	TIL, HUM
5	Fi11-CHP-CCL	Late Core Damage	1.23E-5/ry	S2	Charging, CCW	HPI	TIL, HUM
6	Fi10-CCL	Late Core Damage	1.36E-5/ry	S2	CCW	SSMU, HPI	TIL
7	Fi6-CCL	Late Core Damage	9.65E-6/ry	S2	CCW	SSMU, HPI	TIL
8	Fi8-CCL	Late Core Damage	3.13E-6/ry	S2	Electric Power, CCW		TIL, HUM
9	Fi6-AF3-HR1	Late Core Damage	9.83E-7/ry	T-RX		SSMU, HPR	TIL

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

Loss Supports: At most two of the following: AC, ACBU1, ACBU2, ACBU3, AUXC2, AUXC3, AUXC4, CCW, DC, EAC, EDC, ESAS1, ESAS2, ESW, HVAC1, HVAC2, HVAC3, IA, NIT, OA3, OA4, SA, STM, SW2, SW3, SW4, VAC (Field may be blank).

Failed Functions: At most three of the following: SINT, SDEP, SSMU, 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, TIL, IND-SGTR, SBO, OR HUM (Field may be blank)

Table 5.6
PWR Accident Sequence Detailed Table

Plant Name: Kewaunee

For Fire PRA Only

1 Sheet of 1

#	SEQUENCE	RX		PRIMARY INTEGRITY					PRIMARY INVENTORY-INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT										NOTES
		RPS	BI	PPORV	PSRV	PAD1	PAD2	RCPS	CHP1	HPI	LPI	ACC	AI1	AI2	CHPR	HPR	LPR	AR1	AR2	SGS	SGA	TT	MSIV	TB	SG	MFW	NISP	AFW	AM1	AM2	AM3	CS1	CS2	FC1	FC2	IC	CI1	CI2	IGN	RF	HUM	
1	Fi6-AF3-OB5			X						X																X		X													X	Loss of power
2	Fi4-CHP-CCL							X		X	X					X	X																								X	Loss of power
3	Fi7-AF3			X						X																X		X													X	Loss of offsite
4	Fi2-AF3								X	X	X	X			X	X	X								X	X		X				X	X	X	X						X	Loss of power
5	Fi11-CHP-CCL								X	X	X				X	X	X									X		X				X	X	X	X							Loss of power
6	Fi10-CCL								X	X	X				X	X	X									X		X				X	X	X	X						X	Loss of power
7	Fi6-CCL							X		X	X					X	X									X		X														Loss of CCW
8	Fi8-CCL							X		X	X					X	X																								X	Loss of power
9	Fi6-AF3-HR1															X	X									X		X														

6. REFERENCES

1. "Individual Plant Examination of External Events - Summary Report," Kewaunee Nuclear Power Plant, Wisconsin Public Service Corporation, June 28, 1994.

2. "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities - 10CFR50.54(f)," U. S. Nuclear Regulatory Commission, Generic Letter 88-20, Supplement 4, June 28, 1991.

3. J. T. Chen, et al., "Procedure and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," U.S. Nuclear Regulatory Commission, NUREG-1407, May 1991.

4. "Procedures for the External Event Core Damage Frequency Analysis for NUREG-1150," U.S. Nuclear Regulatory Commission, NUREG/CR-4840, November 1990.

5. "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Electric Power Research Institute, EPRI-NP-6041-SL, Revision 1, August 1991.

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

7. Kewaunee IPE Submittal Report, December 1, 1992.

8. J. W. Reed and R. P. Kennedy, "Methodology for Developing Seismic Fragilities," Electric Power Research Institute, EPRI TR-103959, June 1994.

9. R. T. Sewell, et al., "Individual Plant Examination for External Events: Review Guidance," ERI/NRC 94-501 (Draft), May 1994.

10. "IPEEE Step 1 Review Guidance Document," U.S. Nuclear Regulatory Commission, June 18, 1992.

11. S. C. Lu, and A. Boissonnade, "IPEEE Database Data Entry Sheet Package," Lawrence Livermore National Laboratory, December 14, 1993.

12. "Response to Request for Additional Information Regarding Individual Plant Examination for External Events Submittal," letter to U.S. Nuclear Regulatory Commission, from M. L. Marchi, Wisconsin Public Service Corporation, August 29, 1995.

13. Letter to U.S. Nuclear Regulatory Commission, from Clark R. Steinhardt, Wisconsin Public Service Corporation, October 13, 1995.

14. Personal communication regarding the cable contents in Fire Zone TC-100 between Mr. John Chen of U. S. Nuclear Regulatory Commission and Mr. Alan S. Kuritzky of Energy Research, Inc., October 27, 1995.

15. "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," U.S. Nuclear Regulatory Commission, NUREG/CR-0098, May 1978.

16. "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants (Final Report)," American Nuclear Society and Institute of Electrical and Electronics Engineers, NUREG/CR-2300, Vols. 1 & 2, January 1983.

17. "Evaluation of External Hazards to Nuclear Power Plants in the United States, Other External Events," Lawrence Livermore National Laboratory, NUREG/CR-5042, Supplement 2, February 1989.

18. "Historical Extreme Winds for the United States--Great Lakes and Adjacent Regions," U.S. Nuclear Regulatory Commission, NUREG/CR-2890, August 1982.

19. "Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop Pressurized Water Reactor," U.S. Nuclear Regulatory Commission, NUREG/CR-4458, March 1987.

20. "Updated Safety Analysis Report," Kewaunee Nuclear Power Plant, Revision, July 1, 1993.

21. H.C.S. Thom, "Tornado Probabilities," *Monthly Weather Review*, Vol. 91, No. 10-12, pp. 730-736.

22. E. H. Markee, Jr., et al., "Technical Basis for Interim Regional Tornado Criteria," U. S. Atomic Energy Commission, Office of Regulation, WASH-1300, May 1974.

23. G. E. Lear, and O. O. Thompson, "NRC Inventory of Dams," U.S. Nuclear Regulatory Commission, NUREG-0965, 1983.

24. "Clarification of TMI Action Plan Requirements, Item III.D.3.4, Control Room Habitability," U.S. Nuclear Regulatory Commission, NUREG/CR-0737, November 1980.

25. "Staff Guidance of IPEEE Submittal Review on Resolution of Generic or Unresolved Safety Issues (GSI/USI)," U.S. Nuclear Regulatory Commission, August 21, 1997.