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MEMORANDUM TO: John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Thomas L. King, Director
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

SUBJECT: REVIEW OF INDIAN POINT UNIT 2 INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL

Attached is RES's Staff Evaluation Report (SER) on its review of the Indian Point Unit 2 IPEEE submittal. Also included with the SER are the contractor's Technical Evaluation Report (TER) and a staff TER on the internal floods analysis, which the licensee submitted as part of their IPEEE. We recommend that the enclosed report be formally issued to document the staff's findings and conclusions.

A Step 1 review was performed which examined the IPEEE results for their completeness and reasonableness considering the design and operation of the plant. On the basis of this review and further review by a senior review board (SRB), the staff concluded that the aspects of seismic; fires; and high winds, floods, transportation and other (HFO) external events were adequately addressed. The SRB is comprised of RES and NRR staff and RES consultants (Sandia National Laboratories) with probabilistic risk assessment expertise for external events. The staff's review findings are summarized in the attached SER, and the details of the contractor's findings in the TER appear in an attachment to the SER.

The licensee estimated a mean seismic core damage frequency (CDF) of 1.5E-5 per year using the LLNL seismic hazard curves, and a value about 10% lower using the EPRI seismic hazard curves, before plant modifications. The licensee implemented a modification to the component cooling water (CCW) surge tank supports which lowered the mean seismic CDF to 1.1E-5 per year. The fire CDF was estimated as 1.8E-5 per year. For high winds, floods, transportation and other (HFO) external events, the events other than high winds were either qualitatively or quantitatively screened. For high winds, the core damage frequency was 3.0E-5 per year; tornadoes contributed 1.7E-5 per year, extratropical cyclones 1.1E-5 per year, and hurricane events 2.4E-6 per year to the high winds core damage frequency. The licensee calculated that the contribution to the core damage frequency from internal flooding was 6.7E-6 per year. The CDF due to internal events was estimated to be 3.1E-5 per year, in the licensee's IPE.

The licensee noted, in Section 9.3 of the IPEEE submittal, that a concise definition of "vulnerability" was not provided in the NRC documentation associated with the performance and reporting of the IPEEE. The licensee's IPEEE submittal categorized and evaluated the external event-induced sequences in accordance with the guidelines provided in the Nuclear Energy Institute (NEI) Severe Accident Closure Guidelines NEI 91-04. The licensee identified no vulnerabilities associated with external events, but, with regard to seismic events, the improvement already noted for the CCW surge tank supports was implemented. For external flooding, although no vulnerabilities were identified, some plant improvements associated with the probable maximum precipitation event were identified and either already implemented, or were intended to be implemented, at the time of the IPEEE submittal, as discussed in the attached SER. No needed plant improvements were identified for fire or internal flooding.

As a part of the IPEEE, unresolved safety issue (USI) USI A-45, "Shutdown Decay Heat Removal Requirements," and generic safety issues (GSIs) GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," GSI-103, "Design for Probable Maximum Precipitation (PMP)," GSI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used In Westinghouse Plants," and the Sandia Fire Risk Scoping Study (FRSS) issues were specifically identified during the initial planning of the IPEEE program and explicitly discussed in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407 as needing to be addressed in the IPEEE. The specific information associated with each issue is identified and discussed in the attached SER. Based on the review of the information contained in the submittal, the staff believes that the licensee's process is capable of identifying potential vulnerabilities associated with USI A-45, GSI-57, GSI-103, GSI-131. As far as the FRSS issues are concerned, the staff believes that the licensee's process is capable of identifying potential vulnerabilities, except for the issue of equipment damage caused by operators misdirecting manual fire suppression actions because of smoke; this issue is not addressed in the IPEEE submittal. Misdirection of manual suppression efforts is also part of GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness." Except for the FRSS issue of misdirected manual fire suppression, all of these issues called out directly in Supplement 4 to GL 88-20 and its associated guidance document are considered resolved, on the basis that the process used by the licensee to identify vulnerabilities associated with these issues is judged to be capable of identifying any potential vulnerabilities, and the licensee found no vulnerabilities.

On the basis of the Step 1 review, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the Indian Point Unit 2 IPEEE has met the intent of Supplement 4 to Generic Letter 88-20.

In addition, the licensee's IPEEE submittal contains some specific information that addresses the external event aspects of certain generic issues: GSI-147, "Fire-Induced Alternate Shutdown/Control Room Panel Interactions;" GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness" (also a FRSS issue, mentioned above); GSI-156, "Systematic Evaluation Program (SEP);" and GSI-172, "Multiple System Responses Program (MSRP). The specific information associated with each issue is identified and discussed in the attached SER. Apart from the GSI-148 issue of misdirection of manual fire suppression, the staff considers that the licensee's process for the analysis of these issues is capable of identifying potential vulnerabilities associated with these issues. Since no vulnerabilities associated with the external events aspects of these issues were found, the staff considers these issues resolved for Indian Point 2, except for the GSI-148 issue of misdirection of manual fire suppression. The need for any additional assessment or actions related to the resolution of this issue will be addressed by the NRC staff separately from the IPEEE program.

If you have any questions regarding the attached SER, please contact Arthur Buslik (415-6184). When the SER is issued to the licensee, please put the following staff on distribution: Arthur Buslik, RES; Alan Rubin, RES; Carolyn Woods, RES; and Doug Coe, NRR

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ATTACHMENT

STAFF EVALUATION REPORT

OF

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL

ON

INDIAN POINT UNIT 2 NUCLEAR GENERATING STATION

STAFF EVALUATION REPORT OF INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE) SUBMITTAL ON INDIAN POINT UNIT 2 NUCLEAR GENERATING STATION

I. INTRODUCTION

On June 28, 1991, the NRC issued Generic Letter 88-20, Supplement 4 (with NUREG-1407, Procedural and Submittal Guidance) requesting all licensees to perform individual plant examinations of external events (IPEEE) to identify plant-specific vulnerabilities to severe accidents and to report the results to the Commission together with any licensee-determined improvements and corrective actions. In a letter dated December 6, 1995, the licensee, Consolidated Edison Company of New York, submitted its response to NRC.

The staff contracted with Energy Research, Inc. (ERI) to conduct a "Step 1" review (a review for completeness and reasonableness) of the licensee's IPEEE submittal and its associated documentation and sent a request for additional information (RAI) to the licensee on September 22, 1997. The licensee responded to the RAI on February 24, 1998. Based on the results of the review, the staff concluded that the aspects of seismic; fires; and high winds, floods, transportation and other external events were adequately addressed. However, the Fire Risk Scoping Study issue associated with misdirected manual fire suppression failing equipment was not addressed. This issue is also part of the generic safety issue (GSI), GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness."

The review findings are summarized in the evaluation section below. Details of the contractor's findings are in the technical evaluation report attached to this staff evaluation report. In addition, in a separate attachment, there is a technical evaluation report on the internal floods analysis, which the licensee submitted as part of its IPEEE.

In accordance with Supplement 4 to GL 88-20, the licensee has provided information on the Fire Risk Scoping Study (FRSS) issues, generic safety issue (GSI)-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment," GSI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used In Westinghouse Plants," GSI-103, "Design for Probable Maximum Precipitation (PMP)," and Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements." This information was explicitly requested in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407. The licensee regards these issues as resolved and also considers as resolved the following issues: unresolved safety issue (USI) A-17, "System Interactions at Nuclear Power Plants;" USI A-40, "Seismic Capability of Large Safety-Related Above-Ground Tanks;" and the Eastern US Seismicity (Charleston Earthquake) Issue.

II. EVALUATION

Indian Point Unit 2 is a Westinghouse four-loop pressurized water reactor (PWR) with a large dry containment. The plant is located on the east bank of the Hudson River, in upper Westchester County, about 24 miles from the New York City boundary line. The owner/operator is Consolidated Edison Company of New York. The operating license was issued on September 28, 1973.

Core Damage Frequency Estimates

Seismic

The plant is classified in NUREG-1407 as a 0.3g full scope plant. For the seismic evaluation, the licensee performed a seismic probabilistic risk assessment, with screening based on seismic margin and Seismic Qualification Utility Group methods. The mean estimate of the seismic core damage frequency (CDF) is given as 1.46E-5 per year by the licensee, with the LLNL hazard curves, and about 10% lower when the EPRI curves are used. The licensee made a modification to the component cooling water (CCW) surge tank supports; after the modification the licensee's mean estimate of the seismic core damage frequency was reduced to 1.1E-5 per year.

Fire

For fire, the licensee used the Electric Power Research Institute's fire-induced vulnerability evaluation (FIVE) methodology for both qualitative and quantitative screening, and used probabilistic risk assessment (PRA) methodology for the detailed evaluation. The total fire CDF from the scenarios surviving screening is estimated at 1.8E-5 per year.

High Winds, Floods, Transportation, and Other (HFO) External Events

HFO events, except for high winds and tornadoes, were screened out using the screening criteria given in NUREG-1407. For high winds and tornadoes, a PRA was performed. The contribution to the CDF from tornadoes and extratropical cyclones was estimated by the licensee as 1.7E-5 per year and 1.1E-5 per year, respectively. Hurricane events were estimated to contribute 2.4E-6 per year to the CDF.

Internal Flooding

The licensee's flooding analysis included a screening analysis, and a detailed risk evaluation on nine flood scenarios which survived the screening. The licensee's estimate of the total flood-induced CDF is 6.7E-6 per year.

Dominant Contributors

Seismic

Four seismic damage states (SDSs) contributed about 92% of the seismic CDF. These are:

- Loss of instrumentation and control, due to structural failures of the turbine building frame and the Unit 1 superheater stack
- Loss of CCW, due to failure of the CCW surge tank or failure of the steel superstructure of the fuel storage building
- Loss of 480 VAC electric power, due to seismic failure of cable trays and the seismic failure of 480 VAC motor control centers

- Loss of service water, due to seismic failure of the pumps or heat exchangers, or sliding failure of the intake structure

Fire

The main contributors to the fire CDF are:

- Control room fires, contributing 7E-6 per year to the CDF

- Cable spreading room fires, contributing 4E-6 per year to the CDF

- Switchgear room fires, contributing 4E-6 per year to the CDF

These fires contribute, in total, 1.5E-5 per year to the CDF; this is about 85% of the total fire CDF of 1.8E-5 per year.

HFO

Except for high winds, screening analyses were used. The major HFO contributors to the CDF are tornadoes and extratropical cyclones, contributing 1.7E-5 per year and 1.1E-5 per year, respectively. The dominant high wind core damage sequences are station blackout sequences, responsible for 87% of the high wind CDF. For tornadoes, the dominant structural failures are turbine building failure (leading to consequential failure of the control building), failure of the control building itself, and failure of the diesel generator building together with the gas turbine shelter. The EDG building failures are more important for extratropical cyclones than they are for tornadoes. The reason is evidently that the dominant failure mode for the EDG building is that of suction failure of the roof. If the tornado is not accompanied by rain, the equipment in the EDG building may not fail.

Internal Flooding

The largest contributor to the internal flood CDF was initiated by a break in a 3-inch diameter service water pipe located in the emergency switchgear room. The resulting water flow could not be totally accommodated by the drains, and consequently damage could result in as little as four minutes after the break occurs, assuming no credit for flood detection and isolation due to the limited time available. The second highest contributor was a turbine building flood which resulted in the non-recoverable loss of normal power to the emergency buses due to damage to 6.9 kV buses in the vicinity of the flood. The third highest contributor was a fire protection system pipe break in the deluge valve room located in the control building. The flood propagated to the emergency switchgear room via an interconnecting door.

These three highest contributors to the internal flooding CDF account for 94% of the internal flooding CDF.

Assessment of Licensee's Determination of Dominant Contributors

For seismic, fire, HFO events, and internal flooding, the licensee appears to have identified the significant initiating events and dominant accident sequences.

Containment Performance

Seismic

The licensee did not find, in their IPEEE, any seismic vulnerabilities which could lead to early containment failure or bypass directly as a result of seismic failures of major structures or systems. About 65% of the seismic CDF results in plant damage states with initial loss of containment pressure suppression and heat removal functions. If these functions are not regained, long-term overpressure failure of the containment would likely result, but the conditional probability of early containment failure (e.g., by direct containment heating) would be no more likely than for long term station blackout sequences resulting from internal events initiators.

Fire

Containment performance was evaluated for the potential of fire-induced containment bypass and failure of containment isolation. No significant sequences involving containment bypass or failure of containment isolation were found. The licensee did not evaluate the likelihood of long term containment failure by overpressure.

HFO

HFO events other than high winds were addressed by a screening analysis, consistent with NUREG-1407 guidelines. Although an explicit containment performance analysis is not required for these events, Section 6.2.6 of the IPEEE submittal addresses containment performance for high winds. The section concludes that no vulnerabilities which cause early failures or containment bypass were identified. About 87% of the wind-induced core damage frequency is due to station blackout sequences; in such sequences all containment pressure suppression and heat removal systems are lost. If these functions are not regained, these sequences would likely lead to long term containment overpressure and failure, but the conditional probability of early containment failure (e.g., by direct containment heating) would be no more likely than for long term station blackout sequences resulting from internal events initiators.

Internal Flooding

No explicit discussion of containment performance is given in the IPEEE internal flooding analysis. However, internal flooding sources which result in a loss of primary or secondary reactor coolant outside the containment (for example, interfacing system LOCAs or steam line breaks with failure to isolate) are treated in the Individual Plant Examination (IPE), not the IPEEE. Looking at the dominant contributors to the internal flooding CDF, as given in the IPEEE, it appears that these are equivalent to station blackout core damage sequences. They may lead to long term overpressurization of the containment and containment failure, but the conditional probability of early containment failure (e.g., by direct containment heating), would be no more likely than for long term station blackout sequences resulting from internal events initiators.

Assessment of Licensee's Containment Performance Analysis

The licensee's containment performance analyses for seismic, fire, high winds, and internal flooding events appears to have considered the important severe accident phenomena and are consistent with the intent of Supplement 4 to Generic Letter 88-20.

Generic Safety Issues

As a part of the IPEEE, a set of generic and unresolved safety issues (USI A-45, GSI-131, GSI-103, GSI-57, and the Sandia Fire Risk Scoping Study (FRSS) issues) were identified in Supplement 4 to GL 88-20 and its associated guidance in NUREG-1407 as needing to be addressed in the IPEEE. The staff's evaluation of these issues is provided below.

1. **USI A-45, "Shutdown Decay Heat Removal Requirements"**

   The licensee performed a seismic PRA, a fire PRA, and a PRA for high winds. These are capable of finding vulnerabilities which involve loss of decay heat removal capability. No such vulnerabilities were found. The screening analysis done by the licensee for HFO events other than high winds is capable of finding vulnerabilities associated with loss of decay heat removal capability. Since the staff judges that the process used by the licensee is capable of finding decay heat removal vulnerabilities, and no vulnerabilities were found, the staff considers that the external events aspects of USI A-45 are resolved for Indian Point Unit 2.

2. **GSI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System used in Westinghouse Plants"**

   The flux monitoring cart which is the subject of GSI-131 is seismically robust, with a high confidence low probability of failure (HCLPF) value in excess of 0.5g. (That is, there is 95% confidence that the probability of failure of the cart is less than 5% for a peak ground acceleration of 0.5g). The cart had been previously modified in response to Information Notice 85-45, which had identified the issue.

   Since the IPEEE determined that there was no vulnerability with respect to the flux monitoring cart, using acceptable techniques, the staff considers this issue to be closed for Indian Point Unit 2.

3. **GSI-103, "Design for Probable Maximum Precipitation"**

   The licensee has assessed the effects of the potential for increased plant area flood runoff depth and increased roof loads as a result of the revised Probable Maximum Precipitation (PMP). (See Generic Letter 89-22.) This issue is discussed in Section 6.2.2.3 of NUREG-1407. The staff finds that the licensee's procedure for evaluating

GSI-103 is capable of identifying severe accident sequences resulting from onsite flooding and roof ponding when the revised PMP criteria are used. Some plant improvements were made, as a result of the PMP analysis. These are discussed below, under the heading "Unique Plant Features, Potential Vulnerabilities, and Improvements." On the basis that the licensee's procedure for identifying severe accident sequences associated with the PMP is satisfactory, and on the basis of the improvements that were made, the staff considers that GSI-103 is resolved for Indian Point Unit 2.

4. **GSI-57, "Effects of Fire Protection System Actuation on Safety-Related Equipment"**

The IPEEE submittal, in its discussion of seismic actuation of fire suppression systems (p. 4-43 of the submittal), notes that "fixed fire suppression systems have not been installed where their operation or failure could cause unacceptable damage to safety-related equipment." The submittal also notes, on p. 4-43, that a Con Edison review of Information Notice 83-41 issues concluded that adequate consideration of suppression system actuation effects on safety-related equipment has been integrated into the existing Fire Hazards analysis. The staff finds that the licensee's GSI-57 evaluation is consistent with the guidance provided in EPRI's Fire-Induced Vulnerability Evaluation (FIVE), which was accepted by the NRC staff, and therefore, the staff considers this issue resolved.

5. **Fire Risk Scoping Study (FRSS) Issues**

As noted in the attached TER on the IPEEE submittal, the licensee closely followed the FIVE methodology in addressing the FRSS issues. The FIVE methodology has been accepted by the staff. However, FIVE does not give guidance on the treatment of smoke-induced misdirected manual fire suppression activities, and its potential for failing equipment. The Fire Risk Scoping Study (NUREG/CR-5088) discusses this issue on p.37ff.

The licensee has not provided any discussion in the submittal on smoke-induced misdirection of manual fire suppression efforts. (See Section 2.4.2 of the attached TER.) The staff considers the FRSS issues, except for the effects of misdirected manual fire suppression activities, resolved for Indian Point Unit 2, on the basis that the licensee has used the FIVE methodology for addressing them, and the FIVE methodology has been accepted by the staff.

In addition to those safety issues discussed above that were explicitly requested in Supplement 4 to GL 88-20, four generic safety issues were not specifically identified as issues to be resolved under the IPEEE program; thus, they were not explicitly discussed in Supplement 4 to GL 88-20 or NUREG-1407. However, subsequent to the issuance of the generic letter, the NRC evaluated the scope and the specific information requested in the generic letter and the associated IPEEE guidance, and concluded that the plant-specific analyses being requested in the IPEEE program could also be used, through a satisfactory IPEEE submittal review, to resolve the external event aspects of these four safety issues. The following discussions summarize the staff's evaluation of these safety issues at Indian Point Unit 2:

1. **GSI-147, "Fire-Induced Alternate Shutdown/Control Room Panel Interactions"**

   The licensee has followed the guidance provided in FIVE concerning control systems interactions. Additional details are provided in Section 2.4.1 of the attached TER on the IPEEE submittal.

   Because the FIVE methodology has been accepted by the staff, the analysis of this issue by the licensee is considered acceptable, and the staff considers this issue resolved.

2. **GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness"**

   As noted above, under the discussion of the FRSS issues, the particular aspect of this issue associated with misdirected manual fire suppression because of smoke-induced loss of visibility was not addressed by the licensee in the IPEEE, and is not resolved. Other aspects of operator action effectiveness are addressed (see Section 4.8.5.3 of the IPEEE submittal), and the hindering of short-term (less than 4 hours) operator recovery actions were considered (see the attached TER, Section 2.4.2)

3. **GSI-156, "Systematic Evaluation Program (SEP)"**

   The SEP issues are a set of issues associated with plants that were licensed prior to the time the 1975 Standard Review Plan was issued.

- Settlement of Foundations and Buried Equipment

  The Indian Point Unit 2 site is a rock site, and there are no foundation settlement concerns. As noted in Section 2.4.3 of the attached TER on the IPEEE, buried equipment is not expected to be of concern.

- Dam Integrity and Site Flooding

  The 1982-1983 Indian Point Probabilistic Safety Study evaluated the frequency of probable maximum precipitation and failure of an upstream dam leading to flooding at the plant site to be less than 1E-8 per year. The attached TER notes that the IPEEE submittal does not discuss seismically-induced dam failure. However, during the licensing of Indian Point Unit 3, as noted in Section 6.3 of the Indian Point Unit 2 IPEEE submittal, an analysis concluded that the maximum sustained water surface elevation at the plant is 14.0 feet based on the combined effect of a Hudson River maximum flood, probable maximum precipitation over the Esopus Creek Basin resulting in failure of the Ashokan Dam, and a hurricane at New York Bay. The minimum critical flood height at Indian Point Unit 2 is at 15 feet 6 inches, so that the analyzed flood does not threaten the plant. The event analyzed bounds the seismically-induced failure of the Ashokan Dam.

- Site Hydrology and Ability to Withstand Floods

  The Indian Point IPEEE submittal includes a satisfactory screening analysis of external floods, consistent with NUREG-1407 guidelines, and also satisfactorily analyzed the PMP event (see GSI-103 discussion, above).

- Industrial Hazards

  The IPEEE contains, in its HFO analysis, a satisfactory treatment of these hazards.

- Tornado Missiles

  The effects of tornado missiles were satisfactorily considered in the HFO analysis.

- Severe Weather Effects on Structures

  The effects of high winds and floods were satisfactorily analyzed in the HFO section of the IPEEE.

- Design Codes, Criteria, and Load Combinations

  Since the IPEEE presents a satisfactory analysis of seismic and HFO events, and found no vulnerabilities, it can be inferred that the Category I structures have adequate capacity.

- Seismic Design of Structures, Systems, and Components (SSCs)

  Since the IPEEE presents a satisfactory analysis of seismic and HFO events, and found no vulnerabilities, it can be inferred that the seismic design of SSCs is adequate.

- Shutdown Systems and Electrical Instrumentation and Control Features

  A satisfactory IPEEE analysis, as was done by the licensee for Indian Point Unit 2, automatically includes the study of systems required to remove decay heat, and the instrumentation and control systems required for safe shutdown.

Based on the overall results of the IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with GSI-156. Although seismically-induced dam failures were not explicitly considered, a scenario which bounds the effects of seismically induced failure of the Ashokan dam was considered. On the basis that no potential vulnerability associated with these issues was identified in the IPEEE submittal, the staff considers the IPEEE-related aspects of these issues resolved.

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

- Effects of fire protection system actuation on safety-related equipment

This is issue GSI-57, and is discussed under that heading. See also the attached TER, Section 2.4.4.

- Seismically induced fire suppression system actuations

  This is a FRSS issue, and is discussed as such in Section 4.8.2 of the IPEEE submittal.

- Seismically induced fires

  This is a FRSS issue. Seismically induced fires were addressed in the seismic capability walkdowns performed as part of the seismic IPEEE; the walkdowns are discussed in Section 3.1.3 of the IPEEE submittal. They were also discussed in Section 4.8.2 of the submittal, as part of the discussion of the FRSS issues.

- Effects of hydrogen line ruptures

  The effects of earthquakes on gas lines is addressed in the seismic capability walkdowns, discussed in Section 3.1.3 of the IPEEE, and, in particular, in the discussion of seismic/fire interactions. Hydrogen fire sources are discussed in Section 4.3.2.2 of the submittal (see, in particular p. 4-25 of the submittal).

- *The IPEEE-related aspects of common cause failures associated with human errors*

  With respect to fire, the impact of fires on human actions after a fire initiator is addressed by the IPEEE, in Section 4.6.1.2. Human errors were included in the seismic PRA models (see Section 3.1.6.3 of the IPEEE), and contributed less than 5% to the seismic CDF. As for HFO events, screening analyses were used for all HFO events other than high winds, and no assessment of human error probability was required. For high winds, the methodology developed "wind damage states" which acted as entries to IPE internal events models. These IPE internal events models included the effects of human error.

- Non-safety-related control system/safety-related system dependencies

  As far as the IPEEE is concerned, this issue reduces to that of seismically induced spatial and functional interactions, a MSRP issue already discussed above, and GSI-147, on fire-induced alternate shutdown and control room panel interactions, which has also already been discussed.

- Effects of flooding and/or moisture intrusion on non-safety related and safety-related equipment

  Flooding from external floods is discussed in the HFO analysis in the IPEEE; a screening analysis was used. Flooding from the actuations of fire protection systems is a GSI-57 issue, and is discussed under that heading. Internal flooding is discussed in Chapter 5 of the IPEEE submittal, and the internal flooding core damage frequency, dominant sequences and containment performance are discussed above.

- Seismically induced spatial and functional interactions

  Seismically induced spatial interactions were addressed in the seismic capability walkdowns performed as part of the Indian Point Unit 2 seismic IPEEE, discussed in Section 3.1.3 of the submittal. Seismic functional interactions are addressed as part of the seismic PRA process used by the licensee.

- Seismically induced flooding

  Seismically induced flooding was addressed in the seismic capability walkdowns performed as part of the seismic IPEEE, and is discussed in Section 3.1.3 of the submittal.

- Seismically-induced relay chatter

  Seismically-induced relay chatter was addressed in Section 3.3 of the IPEEE.

- Evaluation of earthquakes greater than the SSE

  The seismic analysis in the IPEEE was a PRA, which automatically includes the effects of earthquakes greater than the SSE.

Based on the overall results of the IPEEE submittal review, the staff considers that the licensee's process is capable of identifying potential vulnerabilities associated with GSI-172. On the basis that no potential vulnerability associated with these issues was identified in the IPEEE submittal, the staff considers the IPEEE-related aspects of these issues resolved.

Unique Plant Features, Potential Vulnerabilities, and Improvements

Unique safety features are described in Section 8.1 of the IPEEE. Those which specifically refer to external events are:

- The Alternate Safe Shutdown System was modified as a result of the original Indian Point Probabilistic Safety Study to more quickly and easily allow power to be provided to key shutdown equipment using power sources which bypass the Indian Point Unit 2 control building areas which contain those buses. This capability is not only useful for fire sequences but also other events such as flooding which may threaten the 480 V buses.

- In addition to the three emergency diesel generators, Indian Point Unit 2 has three gas turbine generators. Since two of the three gas turbines are located some distance from the site, they represent an additional recovery potential for some localized tornado sequences.

The IPEEE used the guidelines provided in the Nuclear Energy Institute (NEI) Severe Accident Closure Guidelines NEI 91-04 to determine whether there were any vulnerabilities which merited physical modification or immediate procedural changes. The IPEEE did not identify any vulnerabilities. However, some improvements in the seismic and HFO areas were made. In particular:

- The hold down bolts for the component cooling water surge tank were replaced by higher tensile strength bolts, reducing the estimated seismic core damage frequency by 29%.

- For the probable maximum precipitation event, a drain flapper valve, located in the manhole to which the control building drains flow, has been added to the preventive maintenance surveillance inspection program.

- In addition, for the probable maximum precipitation event, weather stripping was to be added to the doors leading into the switchgear room from the transformer area to reduce the bottom door gap, and screens are being placed on the equipment hub drains located in the 480V switchgear room to preclude foreign material intrusion. Also, a drain flapper valve, located in the manhole to which the control building drains flow, has been added to the preventive maintenance surveillance inspection program.

III. CONCLUSIONS

On the basis of the overall review findings, the staff concludes that: (1) the licensee's IPEEE is complete with regard to the information requested by Supplement 4 to Generic Letter 88-20 (and associated guidance in NUREG-1407), and (2) the IPEEE results are reasonable given the Indian Point Unit 2 design, operation, and history. Therefore, the staff concludes that the licensee's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities, and therefore, that the Indian Point Unit 2 IPEEE has met the intent of Supplement 4 to Generic Letter 88-20 and the resolution of specific generic safety issues discussed in this SER, with the exception of the FRSS issue associated with equipment damage caused by operators misdirecting manual fire suppression actions because of smoke. This is also part of GSI-148, "Smoke Control and Manual Fire Fighting Effectiveness." The need for any additional assessment or actions related to the resolution of this FRSS issue and GSI-148 will be addressed by the NRC staff separately from the IPEEE program.

It should be noted that the staff focused its review primarily on the licensee's ability to examine Indian Point Unit 2 for severe accident vulnerabilities. Although certain aspects of the IPEEE were explored in more detail than others, the review was not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that underlie or stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPEEE material for purposes other than those associated with meeting the intent of Supplement 4 to GL 88-20 and the resolution of specific generic safety issues discussed in this SER.

Attachment 1

INDIAN POINT UNIT 2

INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS (IPEEE)

TECHNICAL EVALUATION REPORT

ERI/NRC 97-508

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

FINAL REPORT

October 1998

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EXECUTIVE SUMMARY

This technical evaluation report (TER) documents a "submittal-only" review of the individual plant examination of external events (IPEEE) conducted for the Indian Point Unit 2 Nuclear Generating Station (IP2). 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 IP2 IPEEE submittal, particularly with respect to the objectives described in Generic Letter (GL) 88-20, Supplement No. 4, and the guidance presented in NUREG-1407.

The IP2 IPEEE was conducted by Consolidated Edison Company of New York, with some contractor assistance. The IP2 IPEEE submittal considers seismic; fire; and high winds, floods and other (HFO) initiators for the external events analysis. The seismic IPEEE process was a seismic probabilistic risk assessment (SPRA), with screening based on seismic margin and Seismic Qualification Utility Group (SQUG) methods; the fire IPEEE was based on the Electric Power Research Institute's (EPRI's) fire-induced vulnerability evaluation (FIVE) methodology for screening and on probabilistic risk assessment (PRA) methodology for detailed evaluation; and HFO events were evaluated using the screening approach from NUREG-1407 and GL 88-20, Supplement 4, and a PRA for high winds/tornadoes. The IP2 IPEEE was performed in accordance with quality assurance procedures. In addition, the submittal notes that all portions of the IPEEE received several levels of review.

Licensee's IPEEE Process

For the IP2 IPEEE, the licensee had previously completed the Indian Point Probabilistic Safety Study (IPPSS), a Level-3 PRA, in the 1982-1983 time frame. That study was the subject of a detailed technical review by the NRC and Sandia National Laboratories (SNL), as well as the subject of an extensive adjudicatory hearing before a special NRC licensing board panel. In 1989, the licensee updated the Level-1 plant model to reflect changes in systems, equipment, and procedures since completion of the IPPSS. The data analysis was also updated. In 1992, the licensee submitted its Individual Plant Examination (IPE) of internal events, which included model improvements and accounted for plant changes since 1989. The IPEEE submittal builds on these earlier efforts.

The NRC binned IP2 as a 0.3g full-scope plant. The licensee elected to perform a Level-1 SPRA, with a qualitative seismic containment performance analysis. The SPRA generally follows the overall methodology described in NUREG-1407. Plant seismic walkdowns were conducted using the procedures described in

EPRI NP-6041-SL and in the Generic Implementation Procedure (GIP), with screening based on a high confidence of low probability of failure (HCLPF) of 0.5g peak ground acceleration. Walkdown efforts were coordinated for evaluations pertaining to the IPEEE and to Unresolved Safety Issue (USI) A-46. Seismic Evaluation Work Sheets (SEWSs) were completed as part of the equipment reviews. A seismic event tree was developed to model strictly seismic failures, and this tree was then integrated with the existing IPE Level-1 logic models (slightly modified) to incorporate the effects of random (non-seismic) failures and human actions. The study also includes evaluations of relay chatter, seismic-fire interactions, and applicable generic issues (GIs) and USIs. The IPEEE freeze date for plant seismic configuration and procedures was November 1994.

As regards fire-related initiators, the licensee has used FIVE methodology for both qualitative and quantitative screening, and PRA methodology to determine the core damage frequency (CDF). The fire scenarios evaluated as part of the PRA were identified source by source. COMPBRN and FIVE methods were used to determine equipment damage, time to damage, and time to suppression. Fire source frequencies in the FIVE manual were used. The IPE model was modified to reflect the calculated equipment damage of each scenario. The RISKMAN code was used to combine the fire scenario with the plant response to obtain a CDF. Operator actions and post-fire recovery actions were included in the model using the Human Cognitive Reliability (HCR) method. Special sensitivity studies for hot shorts, propagation of fire from high voltage cabinets, control room abandonment probability, and cross zone fire spread were included. Several plant walkdowns were performed during the conduct of the study to obtain data and verify documented data for the screening and PRA phases, as well as for the Fire Risk Scoping Study (FRSS) issues.

The IP2 HFO events IPEEE submittal is based primarily on the screening approach described in Supplement 4 to Generic Letter 88-20. The examination process involved: (1) a review of plant-specific hazard data and plant licensing-basis information; (2) implementation of a qualitative and quantitative screening process; and, (3) a PRA analysis for high winds/tornadoes.

Key IPEEE Findings

The mean seismic CDF was reported as 1.46×10^{-5}/ry for the Lawrence Livermore National Laboratory (LLNL) hazard input; with EPRI seismic hazard input, the mean CDF was about 10% lower. One modification implemented on the component cooling water (CCW) surge tank supports lowered the mean seismic CDF to 1.1×10^{-5}/ry. The main contributors to the seismic CDF are four scenarios involving interaction of the turbine building or Unit 1 superheater stack with the control building. These potential structural failures would lead to loss of instrumentation and control or loss of CCW, and correlated equipment failures leading to loss of alternating current (AC) power or loss of service water. Non-seismic failures and human actions accounted for less than 5% of the seismic CDF, which included the effect of increased human error rates after the occurrence of earthquakes.

For the fire IPEEE, the licensee has concluded that there are no significant fire vulnerabilities at IP2. The total fire CDF from "unscreened" scenarios is estimated to be 1.8×10^{-5}/ry. The main contributors to the fire CDF are the control room, cable spreading room, and a switchgear room. The frequency of core damage and findings of critical plant areas are typical of results from similar nuclear power plants (NPPs).

For HFO events, with the exception of high winds/tornadoes, all other initiators were either qualitatively or quantitatively screened. The contribution to CDF from tornadoes and extratropical cyclones is 1.7×10^{-5}/ry and 1.1×10^{-5}/ry, respectively. Hurricane events contribute 2.4×10^{-6}/ry.

Generic Issues and Unresolved Safety Issues

The seismic IPEEE addressed USI A-45, "Shutdown Decay Heat Removal Requirements" and GI-131, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants." The seismic IPEEE also addressed the Charleston Earthquake Issue and coordination efforts with USI A-46, "Verification of Seismic Adequacy of Equipment in Operating Plants," as well as the seismic spatial interactions aspects of USI A-17, "Systems Interactions in Nuclear Power Plants," and USI A-40, "Seismic Design Criteria: Seismic Capability of Large Safety-Related Above-Ground Tanks." For USI A-45, no vulnerabilities were identified related to decay heat removal. For GI 131, seismic upgrades had previously been implemented for the flux monitoring cart, and the seismic IPEEE fragility analysis screened these components as high capacity components.

For the fire IPEEE, all of the Sandia Fire Risk Scoping Study (FRSS) issues and USI A-45 issues have been addressed. For both cases, the licensee has concluded that there are no outstanding problem areas. The licensee has presented discussion for each issue, and plant walkdowns have been undertaken to assist in resolution of these issues. Seismic-fire interaction issues were addressed by examination of the potential for an earthquake-induced fire event, for inadvertent seismic actuation of the fire suppression system and resulting adverse effects on safety equipment, and for seismic-induced failure of the fire protection systems. The licensee concluded that there are no areas in the plant where inadvertent actuation of fire suppression systems could lead to safety equipment damage. Specific inspection and testing procedures have been instituted to verify the integrity of penetration seals, fire barriers, and fire dampers. Regarding USI A-45, heat removal capabilities have been addressed in detail in the fire analysis via the IPE models used for CDF evaluation. The model gives credit to the possibility of bleed-and-feed cooling.

The submittal discusses the effects of the probable maximum precipitation (PMP) on the plant (GI-103). The HFO IPEEE submittal does not describe the formal analysis of any other safety issues. It does, however, state that some GIs and USIs were addressed and are considered closed. The submittal considers the following issues to be closed:

- USI A-45, "Shutdown Decay Heat Removal Requirements"
- USI A-17, "System Interactions in Nuclear Plants"
- GI-103, "Design for Probable Maximum Precipitation (PMP)"

Some information is also supplied in the IPEEE submittal which pertains to generic safety issues (GSIs) 147, 148, 156, and 172.

Vulnerabilities and Plant Improvements

No seismic-related vulnerabilities have been identified for IP2. One improvement related to higher tensile strength hold-down bolts for the CCW surge tank was implemented; no other improvement or commitments related to seismic issues were identified in the submittal.

No fire-related vulnerabilities have been identified from the IPEEE. The licensee has used the Nuclear Energy Institute's (NEI's) severe accident closure guidelines (NEI 91-04) to evaluate the need for plant improvements. No improvements or commitments were deemed necessary to further reduce the fire risk at IP2.

The IPEEE for IP2 has identified no vulnerabilities and no needed plant improvements with respect to severe accident risk from any HFO initiators.

Observations

Judged on the basis of the submittal, this review concludes that the licensee's seismic IPEEE methodology is capable of identifying severe accident vulnerabilities. It appears that the licensee understands the plant and seismic PRA techniques, and has conscientiously applied this knowledge to produce the seismic IPEEE submittal. The IP2 seismic IPEEE is comprehensive with respect to the important points of GL 88-20 and NUREG-1407. The strengths of the submittal are its (1) development of a comprehensive equipment list, (2) systematic and comprehensive walkdown, including structure and component screening, and consideration of interaction issues, (3) implementation of seismic PRA technology, and (4) relay chatter evaluation in coordination with its USI A-46 study.

For the evaluation of fire initiators, the licensee demonstrated detailed knowledge of the plant and fire PRA methodology, and has made a conscientious application of this knowledge. The licensee has employed proper methodology (i.e., the EPRI FIVE methodology for screening, and a PRA methodology for CDF quantification), and has employed proper data bases and calculational methods for fire occurrence and suppression system failure rates. The many low contribution scenarios are typical of the ignition source driven PRA method used. Notable strengths of the submittal include: (1) assumptions, sensitivity studies and uncertainties are well presented, (2) the inter-compartment fire propagation analysis was unusually thorough, and (3) the hot short analysis was far more comprehensive than is typical for an IPEEE. The final conclusions of the submittal are reasonable, and are within the range of results expected for a pressurized water reactor (PWR). The licensee's fire IPEEE process is capable of identifying severe accident vulnerabilities and none were found.

The HFO evaluation implemented the progressive screening method of NUREG-1407. Because IP2 generally does not meet the SRP, additional analyses were performed, as needed. Hazard screening and verification walkdowns were done appropriately, and changes since issuance of the OL were noted. Good use was made of earlier PRA work. The state-of-the-art wind PRA is particularly noteworthy. The analysis was comprehensive per NUREG-1407. No significant weaknesses were noted during this review

PREFACE

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

Seismic

S. Sholly and M. Frank

Fire

J. Lambright and M. Frank

High Winds, Floods and Other External Events

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Review Oversight, Coordination and Integration

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This work was performed under the auspices of the United States Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. The continued technical guidance and support of various NRC staff is acknowledged.

ABBREVIATIONS

AC	Alternating Current
AFW	Auxiliary Feed Water
ASSS	Alternate Safe Shutdown System
ATWS	Anticipated Transients Without Scram
BOP	Balance of Plant
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCW	Component Cooling Water
CDF	Core Damage Frequency
CST	Condensate Storage Tank
DBE	Design Basis Earthquake
DC	Direct Current
DHR	Decay Heat Removal
EDG/GT	Emergency Diesel Generator/Gas Turbine
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
ERPG-2	Emergency Response Planning Guidelines-2
FIVE	Fire Induced Vulnerability Evaluation Method
FPS	Fire Protection System
FRSS	Fire Risk Scoping Study
GI	Generic Issue
GIP	Generic Implementation Procedure
GL	Generic Letter
GSI	Generic Safety Issue
HCLPF	High Confidence of Low Probability of Failure (capacity)
HCR	Human Cognitive Reliability
HEP	Human Error Probability
HFO	High Winds, Floods and Other (external initiators)
HVAC	Heating, Ventilation and Air Conditioning
IP2	Indian Point Unit 2
IP3	Indian Point Unit 3
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
IPPSS	Indian Point Probabilistic Safety Study
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
MCC	Motor Control Center
MFW	Main Feed Water
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
MSRP	Multiple System Responses Program
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant

NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OL	Operating License
PGA	Peak Ground Acceleration
P&ID	Piping and Instrument Diagram
PMF	Probable Maximum Flood
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
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SCBA	Self-Contained Breathing Apparatus
SDS	Seismic Damage State
SEL	Seismic Equipment List
SEP	Systematic Evaluation Program
SET	Seismic Event Tree
SEWS	Seismic Evaluation Work Sheet
SI	Safety Injection
SLOCA	Small-break Loss of Coolant Accident
SMA	Seismic Margin Assessment
SNL	Sandia National Laboratories
SPRA	Seismic Probabilistic Risk Assessment
SQUG	Seismic Qualification Utility Group
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
TER	Technical Evaluation Report
UE&C	United Engineers and Constructors Corporation
UFSAR	Updated Final Safety Analysis Report
UHS	Uniform Hazard Spectrum
USI	Unresolved Safety Issue

1 INTRODUCTION

This technical evaluation report (TER) documents the results of the "submittal-only" review of the individual plant examination of external events (IPEEE) for the Indian Point Unit 2 Nuclear Generating Station (IP2) [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, Consolidated Edison Company of New York, 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.

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 IP2 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. Section 2.4 identifies the locations in the IPEEE submittal where information having potential relevance to generic safety issues (GSIs) 147, 148, 156 and 172 may be found. 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

IP2 is a four-loop, pressurized water reactor (PWR) with a Westinghouse-designed nuclear steam supply system (NSSS). The unit is rated at 3071.4 Mw thermal and is enclosed in a steel-reinforced, cylindrical, large, dry containment structure. The balance of plant (BOP) systems were designed by United Engineers and Constructors (UE&C) Corporation. The plant site is located on the east bank of the Hudson River, within the village of Buchanan, in upper Westchester County, New York. The site is about 24 miles north of the New York City boundary line. The nearest city is Peekskill, 2.5 miles northeast of IP2.

The major plant structures are the primary auxiliary building, the containment building, the control building, the turbine building, and the emergency diesel generator building. The principal safety systems of IP2 are three safety injection (SI) system pumps (pump shutoff head is below the power operated relief valve [PORV] setpoint), four accumulators, two residual heat removal (RHR) pumps for low pressure injection (and heat exchangers for primary decay heat removal), two low pressure recirculation pumps, two containment spray pumps, five containment fan coolers, four steam generators supplied from two motor-driven auxiliary feed water (AFW) pumps (each feeding two steam generators) and one turbine-driven AFW pump (feeding all four steam generators), three emergency diesel generators feeding four 480 V alternating current (AC) buses, three gas turbine generators with blackstart capability, and four direct current (DC) buses fed from battery chargers or four battery banks. Support systems include component cooling water (CCW) (three pumps and two heat exchangers, cooled by the service water system) and service water systems (both essential and non-essential, with three pumps on each system).

IP2 is co-located on the site with Indian Point Unit 3 (IP3), operated by the New York Power Authority, and the older, decommissioned Unit 1. Co-location with Unit 1 is significant because (a) one of the important seismic sequences occurs because of interaction with the Unit 1 superheater stack and the Unit 2 control building, and (b) the Unit 2 Alternate Safe Shutdown System (ASSS) relies on Unit 1 auxiliary equipment.

1.2 Overview of Licensee's IPEEE Process and Important Insights

1.2.1 Seismic

As documented in NUREG-1407, IP2 is binned into the 0.3g full-scope seismic review category. Consolidated Edison undertook a Level-1 seismic probabilistic risk assessment (SPRA), with a qualitative seismic containment performance assessment, for the IP2 seismic IPEEE. The IPEEE submittal was an update of the Indian Point Probabilistic Safety Study (IPPSS) [4] which was one of the first modern, full-scope probabilistic risk assessments (PRAs) performed in the early 1980s. The seismic IPEEE analysis updated that analysis with a state-of-the-art seismic hazard analysis, a wider scope of initiators, more detailed analysis, and much more detailed seismic walkdowns than were performed for the IPPSS effort.

The specific elements of the IP2 seismic IPEEE, as described in the submittal report, include:

- Hazard Analysis
- Compilation of Seismic Equipment List (SEL)
- Walkdowns (A-46 Generic Implementation Procedure [GIP] and Electric Power Research Institute [EPRI] NP-6041, Rev. 1)
- Analysis of Plant System and Structural Response
- Evaluation of Fragilities and Failure Modes
- Analysis of Plant Systems and Sequences
- Evaluation of Containment Performance
- Low Ruggedness Relay Chatter Evaluation
- Review of Unresolved Safety Issue (USI) A-45, Generic Issue (GI)-131, and Other Seismic Safety Issues

The SPRA methodology followed the guidance in NUREG-1407 and GL 88-20. The SEL considered the A-46 evaluation, and the SEL was used to guide the scope of the walkdowns and low ruggedness relay evaluation; approximately 800 components were placed on the SEL. Walkdowns followed the EPRI seismic margin assessment (SMA) guidelines [5] and the Seismic Qualification Utility Group (SQUG) GIP [6]. A seismic event tree (SET) was developed to identify seismic scenarios which were described as "seismic damage states" (SDSs). The frequencies of these damage states were quantified by convolving the mean hazard curve with the structural and equipment fragility curves. For scenarios requiring random (non-seismic) failures, the individual plant examination (IPE) internal events model was used to develop conditional core damage probabilities (CCDPs). The plant logic models include seismic, random, and human failure events.

To calculate the seismic demand on structures and components, estimates of structural response were scaled from the new design basis earthquake (DBE) spectra developed for the A-46 evaluations. A structural response factor of safety was estimated by comparing the spectral accelerations for the A-46 spectra and damping to the uniform hazard spectrum (UHS) and median centered damping used for the SPRA. This factor and its variability were used to scale design loads and spectra and to define the uncertainty in these loads and spectra. For structures and components that were not screened out based on the seismic capacity walkdowns, progressively more detailed calculations were performed to estimate the seismic capacity. For structures, existing calculations performed for the 1982 IPPSS analysis were updated to reflect the use of the UHS spectral shape for defining ground motion, and to incorporate refinements in the methodology used since the IPPSS analysis was performed. For equipment, a combination of updated IPPSS calculations, generic calculations, and extrapolations of A-46 calculations, was used to determine fragilities. A surrogate fragility (high confidence of low probability of failure [HCLPF] of 0.50g, median fragility of 1.5g, $\beta_R = 0.30$, $\beta_U = 0.36$) was defined for all screened out components. This surrogate fragility, when convolved with the mean seismic hazard, results in a frequency of seismic failure of less than 1.0×10^{-6}/ry.

The SEISMIC code was used to quantify the frequency of SDSs. SEISMIC uses a Monte Carlo sampling process at each seismic magnitude interval to combine the hazard and fragility information for each structure/component in the SDS equation. The code repeats the process for each seismic magnitude interval, and then sums the results to obtain the SDS frequency. The process is repeated for each SDS equation until all equations are quantified.

The seismic core damage frequency (CDF) was evaluated to be 1.46×10^{-5}/ry using Lawrence Livermore National Laboratory (LLNL) seismic hazard input; use of the EPRI seismic hazard input resulted in a 10% reduction in overall seismic CDF. As a result of the IPEEE analysis, it was determined that the capacity of the CCW surge tank was limited by the capacity of the hold down bolts. These hold down bolts were replaced with higher tensile strength bolts, and the seismic model was requantified, resulting in an updated seismic CDF estimate of 1.1×10^{-5}/ry.

Additional details pertaining to the various major aspects of the seismic IPEEE process and findings are provided in Section 2.1.

1.2.2 Fire

The licensee has used FIVE methodology for both qualitative and quantitative screening, and PRA methodology to determine the core damage frequency (CDF). The fire scenarios evaluated as part of the PRA were identified source by source. COMPBRN and FIVE [7] methods were used to determine equipment damage, time to damage, and time to suppression. Fire source frequencies in the FIVE manual were used. The IPE model was modified to reflect the calculated equipment damage of each scenario. The RISKMAN code was used to combine the fire scenario with the plant response to obtain a CDF. Operator actions and post-fire recovery actions were included in the model using the Human Cognitive Reliability (HCR) method. Special sensitivity studies for hot shorts, propagation of fire from high voltage cabinets, control room abandonment probability, and cross zone fire spread were included. Several plant walkdowns were performed during the conduct of the study to obtain data and verify documented data for the screening and PRA phases, as well for the Fire Risk Scoping Study (FRSS) issues [8].

The licensee has assessed the overall fire CDF for "unscreened" scenarios to be 1.8×10^{-5}/ry. The main contributors to the fire CDF are the control room, cable spreading room, and a switchgear room.

The FRSS issues have been considered by closely following the methodology recommended by FIVE. Also considered was the adequacy of decay heat removal (USI-A45). The licensee has concluded in both cases that there are no outstanding problem areas. The licensee has also concluded that there are no significant fire vulnerabilities at IP2, and has used the Nuclear Energy Institute (NEI) severe accident closure guidelines [9] to evaluate the need for plant improvements. No improvements or commitments were identified as being necessary to further reduce the fire risk at IP2.

1.2.3 HFO Events

The licensee has conducted a detailed analysis of HFO events. Most events have been screened utilizing qualitative and quantitative arguments. The contribution to CDF from tornadoes and extratropical cyclones is 1.7×10^{-5}/ry and 1.1×10^{-5}/ry, respectively. Hurricane events contribute 2.4×10^{-6}/ry. Based on an analogy to NEI guidance with respect to severe accident closure, the IP2 IPEEE submittal reports no vulnerabilities with respect to HFO events.

1.3 <u>Overview of Review Process and Activities</u>

In its qualitative review of the IP2 IPEEE, ERI focused on the study's completeness in reference to NUREG-1407 guidance; its ability to achieve the intent and objectives of GL 88-20, Supplement No. 4; its strengths and weaknesses with respect to the state-of-the-art; and the robustness of its conclusions. This review did not emphasize confirmation of numerical accuracy of submittal results; however, any numerical errors that were obvious to the reviewers are noted in the review findings. The review process included the following major activities:

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

Because these activities were performed in the context of a submittal-only review, ERI did not perform a site visit or an audit of either plant configuration or detailed supporting IPEEE analyses and data. Consequently, it is important to note that the ERI review team did not verify whether or not the data presented in the IPEEE matches the actual conditions at the plant, and whether or not the programs or procedures described by the licensee are indeed implemented at IP2.

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* [10], for review of a seismic PRA, and the guidance provided in the NRC report, *IPEEE Step 1 Review Guidance Document* [11]. In addition, on the basis of the IP2 IPEEE submittal, ERI completed data entry tables developed in the LLNL document entitled *IPEEE Database Data Entry Sheet Package* [12].

In its IP2 seismic review, ERI examined Sections 1, 2, 3, 4.8.2, 7, 8, and 9 of the IPEEE submittal for IP2 [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:

1. Were appropriate walkdown procedures implemented, and was the walkdown effort sufficient to accomplish the objectives of the seismic IPEEE?
2. 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?
3. Were component demands assessed in an appropriate manner, using valid seismic motion input and structural response modeling?
4. Were fragility calculations performed for a meaningful set of components, and are the fragility results reasonable?
5. 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?
6. Was the approach to seismic risk quantification appropriate, and are the results meaningful?
7. 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?
8. Has the seismic IPEEE produced meaningful findings, has the licensee proposed valid plant improvements, and have all seismic risk outliers been addressed?

In some instances, quick calculations have been performed as part of the seismic review in order to check the implications of various intermediate and final results.

1.3.2 Fire

During this technical evaluation, ERI reviewed the fire events portion of the IPEEE for completeness and consistency with past experience. This review was based on Sections 1, 2, 4, 5, 7, 8, and 9 of Reference [1]. The guidance provided in References [10,11] was used to formulate the review process and organize this TER. The data entry sheets provided in Section 5 are taken from Reference [12].

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, 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 PRAs. For a few fire zones/areas that were deemed important, ERI also verified the logical development of the screening justifications/arguments (especially in the case of fire-zone screening) and the computations for fire occurrence and CDF.

1.3.3 HFO Events

The review process for HFO events closely followed the guidance provided in Reference [11]. This process involved examinations of the methodology, the data used, and the results and conclusions derived in the submittal. The IPEEE methodology was reviewed for consistency with currently accepted practices and NRC recommended procedures. Special attention was focused on evaluating the adequacy of data used to estimate the frequency of HFO events, and on confirming that any analysis of 1975 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, in some instances, computations of frequencies of occurrence of hazards, fragility values, and failure probabilities were spot checked. Review team experience was relied upon to evaluate the reasonableness of the licensee's evaluation.

2 CONTRACTOR REVIEW FINDINGS

2.1 Seismic

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

2.1.1 Overview and Relevance of the Seismic IPEEE Process

a. Seismic Review Category

As documented in NUREG-1407, IP2 is binned as a 0.3g full-scope plant.

b. Seismic IPEEE Process

The licensee updated the earlier IPPSS analysis to perform the SPRA, together with a qualitative evaluation of containment performance, for the IPEEE of IP2. The SPRA has made use of the IPE, which is also an update of an existing PRA study. The seismic IPEEE was extensively coordinated with the licensee's USI A-46 evaluation, particularly with respect to plant walkdowns and relay ruggedness evaluation. The analysis was a joint utility, NUS, and EQE evaluation, with utility personnel involved in all aspects of the study.

c. Review Findings

The seismic IPEEE methodology is appropriate and relevant to severe accident analysis and vulnerability assessment. The licensee had a meaningful participation in the study.

2.1.2 Logic Models

The plant logic analysis for IP2 includes the following three major aspects: (a) seismic initiating events analysis, (b) event tree development, and (c) fault tree development.

a. Seismic Initiating Events Analysis

The IPE loss of offsite power (LOSP) event tree was used as the initial model for the seismic PRA. Based on guidance in NRC and EPRI seismic margin procedures, large loss of coolant accidents (LOCAs), intermediate LOCAs, main steam line breaks (up to the main steam isolation valves [MSIVs]), main feed water (MFW) line breaks inside containment, and seismically-induced steam generator tube ruptures were screened from further consideration.

An SET was developed and used to identify the potential successes and failures that could occur as a result of a seismic event. Seismically-induced failures incorporated into the SET were obtained from seismic fragility evaluations. Seismically-induced failures of redundant components were assumed to be completely correlated by treating redundant components as though they were a single component in the SET model. The SET includes only seismically-induced impacts. Non-seismic, random failures and operator errors were addressed separately from the SET.

Some components and systems were not included in the SET and were conservatively assumed to be unavailable as a result of an earthquake. Examples of such components and systems include the city water system (which can be used to supply cooling to charging pump seals if CCW is lost), the primary water system (which can be used to supply cooling to the charging pumps), and the three gas turbine generators (which can provide an alternate station power source). In addition, if the superheater stack collapses onto the control building, or the turbine building impacts the control building, causing failure of plant instrumentation and control, it may be possible to control the plant from the Alternate Safe Shutdown System (ASSS). Due to the potential impact on plant operations personnel from such events, however, the ASSS was not credited in the SPRA.

b *Event Tree Modeling*

Boolean equations were developed for each of the SET top events based on the logic and seismic fragility information. The equations represent seismic failures and successes of structures and components. Each seismic sequence equation represents the Boolean logic associated with its corresponding SDS. Using the SEISMIC code, the SET was quantified without fan cooler failures. For SDSs with frequencies greater than 1.0×10^{-7}/ry, the branch point for fan cooler failures was then addressed by adding two equations, one with fan cooler failure and one with fan cooler success. (Some SDSs do not address fan cooler failure because the failure of the fan coolers is guaranteed, e.g., loss of all AC power.)

The internal events IPE model was used to calculate the impact of non-seismic failures and human errors for each of the non-negligible SDSs. Some changes were made to the IPE logic model as follows:

- Emergency diesel generator and fuel oil pump run times were increased from 6 to 24 hours to reflect the increased difficulty of restoring offsite power and making repairs following an earthquake.
- The gas turbines were assumed to fail if offsite power failed (due to relatively low capacity).
- Loss of CCW was assumed to result in a non-recoverable reactor coolant pump (RCP) seal LOCA and consequential core damage (because alternate sources of water such as city water and primary water are assumed to be unavailable after an earthquake).
- City water to the AFW pumps was assumed to be unavailable.
- For SDSs with a frequency less than 1.0×10^{-7}/ry, the fan coolers were conservatively assumed to be unavailable.
- For anticipated transients without scram (ATWS) events, emergency boration and manual scram were assumed to be unavailable.

c. *Fault Tree Modeling*

The fault tree modeling of the SET was handled with Boolean equations representing the combinations of failure events. These combinations were assembled, together with success events, into expressions which were evaluated for the SDSs. Random failures and operator errors are represented in the revised IPE fault trees as basic events.

d. Review Findings

The logic modeling process for the IP2 SPRA appears to be an acceptable approach that is adequate with respect to NUREG-1407 guidelines.

2.1.3 Non-Seismic Failures and Human Actions

The plant logic models incorporate non-seismic failures and human actions in addition to seismic failures. Non-seismic failure rates are the same as those used for the internal events analysis.

Operator error probabilities were adjusted to reflect seismic impacts. For operator actions which can be taken at later than one hour after the seismic initiating event, the human error probability (HEP) from the IPE models is used. For operator actions which have to be taken in less than an hour, HEPs were adjusted to account for the confusion, distraction, and potential difficulty in movement associated with earthquakes. The adjustments were based on the IPE HEP values as follows:

IPE HEP Range	Adjustment Factor (Multiplier)
0.1 to 1.0	2 (maximum HEP of 1.0)
0.01 to 0.1	5 (maximum HEP of 0.2)
<0.01	10

No unique operator errors were added to the model for the seismic PRA. Some operator actions were not possible after an earthquake, and were therefore excluded from the model (e.g., action to use city water).

This review notes that there is no research, data or other technical basis for earthquake related HEP adjustment factors. Nevertheless, the licensee made an effort to judgementally assess the effects of an earthquake on operator action. The values noted above are consistent with adjustment factors used in other seismic IPEEE studies. The licensee reported that non-seismic failure and human actions were not among the most significant contributors to CDF.

In summary, it is judged that non-seismic failures and human actions have been addressed in a manner consistent with the guidelines of NUREG-1407.

2.1.4 Seismic Input (Ground Motion Hazard and Spectral Shape)

The revised LLNL seismic hazard estimates for IP2 (NUREG-1488) were used for the baseline frequency calculations. The IPEEE submittal observed that the EPRI seismic hazard estimates are essentially the same as the revised LLNL estimates.

The seismic hazard was truncated before 1.5g; the explicit mean hazard from NUREG-1488 was used, which extends only to 1.0g. The IPEEE submittal points out that the maximum increase in seismic CDF from extrapolating the hazard to 1.5g or beyond would be 1.75×10^{-6}/ry (which is the exceedance frequency of the 1.0g acceleration). This would contribute about 10% compared to the overall seismic CDF calculated in the study. Examination of the impact of truncating the curve at 1.0g indicated that the ranking of dominant sequences and their dominant contributors remains the same. Thus, no additional insights would be available from extending the hazard curve by extrapolation into the range beyond 1.0g.

The spectral shape used in the IP2 analysis considered the EPRI UHS.

In summary the IPEEE's use of EPRI and LLNL hazard curves is appropriate. Since NUREG-1407 allows the use of the EPRI UHS shape, the seismic input spectrum and ground motion hazard used for the IP2 SPRA are acceptable.

2.1.5 Structural Responses and Component Demands

Structural response for plant structures was originally analyzed during the design stage of IP2. As part of its response to the A-46 issue, the licensee contracted with Altran Corporation to reanalyze the structural response for the design basis earthquake and develop in-structure spectra. No new analyses were performed for the IPEEE; rather, the Altran analysis was scaled to develop loads and spectra for use in the IPEEE. A structural response factor of safety was developed by comparison of the spectral accelerations for the DBE design analysis spectra and damping to the UHS and median centered damping used for the IPEEE. This factor and its variability were used to scale design loads and spectra, and to define the uncertainty in these loads and spectra. A similar scaling process was used for equipment. However, slightly different response factors were used for structure and equipment fragility evaluations, because a time history record was used to generate the in-structure response spectra for equipment, while a response spectrum analysis was conducted for the development of structural loads. The structural response factor for equipment is greater than for structures as a result of the fact that the response spectrum of the time history exceeds the response spectrum used to determine loading in the buildings. The median factor of safety, as well as random variability and uncertainty, were estimated for each of the parameters affecting the capacity and response. These estimates were then combined to obtain an overall median factor of safety and variability estimates. The structural response factors for equipment are summarized in Table 3.1-1 of the submittal.

In summary, the development of structural responses and component demands in the IP2 seismic IPEEE appears to be consistent with the relevant guidelines presented in NUREG-1407.

2.1.6 Screening Criteria

The EPRI SMA screening criteria described in EPRI NP-6041, Rev. 1 [5], define the framework used in making screening decisions. All components meeting the screening criteria and caveats associated with a HCLPF level of 0.5g peak ground acceleration (PGA) were screened out. The submittal also notes that the GIP screening criteria [6] were applied for evaluation of components.

The screening criteria and procedures used in the IP2 seismic IPEEE are consistent with NUREG-1407 guidelines.

2.1.7 Plant Walkdown Process

The components included in the walkdown were those on the SEL. The SEL began with the IPE list of equipment. The following steps were performed to update that list:

- Determine potential initiating events that could occur with a seismic event (either due to the earthquake or due to random or consequential events).

- Determine which safety functions would be required to respond to these initiating events, and which systems provide these safety functions to mitigate the events.

- Remove systems and equipment from the IPE list which are either not required or not available.

- Remove generically rugged, passive components (e.g., check valves and manual valves).

- Add components for pressure boundary integrity.

- Add components for containment performance.

- Add electrical panels, cabinets, and instrument racks.

- Review LOSP emergency procedures, and add equipment and instrumentation which would be needed after an earthquake.

- Add unique IP2 equipment or features.

- Cross-check and verify with the A-46 equipment list.

A seismic systems walkdown (pre-walkdown) was performed to identify any cabinets or equipment that should be included in the SEL, but was not identified from the systems analysis or piping and instrument diagrams (P&IDs); to identify multiple equipment items mounted on a skid; to identify obvious spatial interactions, in order to alert the fragilities analysts; and to identify any seismic/fire or seismic/flood interactions for analysis by fragility experts. Using these steps, about 800 components were placed on the SEL for the seismic walkdowns. All structures housing critical equipment, as well as other structures adjacent to these critical structures that could potentially interact with the critical structures, were added to the SEL.

Many components in the IPEEE walkdown were also on the A-46 Safe Shutdown Equipment List (SSEL). The A-46 SSEL components were walked down prior to the IPEEE seismic capability walkdowns. Because the screening rules for A-46 (per the GIP) are similar to the rules for the seismic margin walkdown under EPRI NP-6041, Rev. 1, the components which are common to A-46 and IPEEE did not require a second detailed walkdown for the IPEEE. The IPEEE seismic capability team reviewed the A-46 equipment data files and did a "walk by" of the equipment to evaluate seismic/fire, seismic/flood, and spatial interactions applicable to beyond DBEs. There were members of the IPEEE walkdown team in common with the A-46 walkdown team who were familiar with the equipment.

The IPEEE submittal describes the expertise collectively present in the walkdown teams, but does not provide details concerning the teams' composition. The submittal contains a summary of the walkdown procedure, but little detail about specific insights gained by the walkdown. The walkdowns were conducted in two parts. Components inside the containment were inspected in February 1993 during a refueling outage, concurrently with the A-46 review. The remaining components were inspected in July 1993, with a small number of follow-up items resolved during the 1995 refueling outage. Photographs were used to supplement the seismic evaluation work sheets (SEWSs) generated during the walkdowns.

In summary, the submittal presents a well-structured description of the walkdown process. A pre-screening of structures was undertaken, followed by a walkdown verification of screening decisions. The approach taken for identifying IPEEE equipment and other PRA components is consistent with applicable guidance. Systems engineers were consulted, from the beginning of the IPEEE process, in developing this list. The IPEEE SEL was cross-walked with the A-46 SSEL to ensure comprehensiveness in the IPEEE evaluation. The submittal's description of plant seismic walkdowns suggests that a well-structured, comprehensive, detailed, and effective approach was employed. The IPEEE walkdown process is thus judged to be relevant to severe accident evaluation of IP2.

2.1.8 Fragility Analysis

Fragilities were not developed for all structures and equipment. Rather, only components estimated to have a median capacity of less than 1.5g PGA, or a HCLPF capacity of less than 0.5g, were considered. Structures and components which were screened out are represented by surrogate fragilities.

The fragility analysis followed the guidance and procedures in EPRI TR-103059 [13]. For structures, the original IPPSS calculations were revised to reflect the use of UHS for defining the ground motion spectral shape, and to incorporate refinements in methodology since the IPPSS calculations were performed. For equipment, a combination of methods was used. Some of the original IPPSS calculations were updated to reflect the use of a UHS. Most components on the SSEL were screened out based on generic calculations, review of A-46 calculations, review of test reports, or by judgment based on the walkdown observations and generic ruggedness. For most cases where fragility calculations had to be performed, the A-46 calculations served as the basis for development of fragilities. Fragility calculations are summarized in Table 3.1-3 of the submittal.

In summary, a logical and efficient methodology has been implemented for fragility analysis of screened-in components. Screened-out components have been assigned a surrogate-element fragility, which is a lower bound capacity derived from seismic margin screening limits and the site-specific spectral shape. Since the surrogate element turns out not to be a dominant contributor in the IP2 IPEEE (and hence, does not mask seismic vulnerabilities), its use appears appropriate. The fragility analysis is judged to be relevant to severe accident evaluation at IP2 and consistent with the guidelines in NUREG-1407.

2.1.9 Accident Frequency Estimates

The mean seismic hazard curve (LLNL, NUREG-1488), structural/equipment fragilities, and SDS equations were input to the SEISMIC code to quantify the frequency of the SDSs. The SEISMIC code uses a Monte Carlo sampling process at each seismic magnitude interval to combine the hazard and fragility information for the SDS equation. The code repeats this process for each seismic magnitude and SDS equation, and sums the results to obtain the SDS frequencies. The core damage and plant damage state frequency from each non-negligible SDS was quantified by modifying the plant logic model in the RISKMAN software to account for the frequency of each SDS and the associated structural or equipment damage.

The base case quantification used the mean LLNL hazard curve, resulting in a mean seismic CDF of 1.46×10^{-5}/ry. Sensitivity results were calculated for several cases. A modification to the CCW surge tank supports was performed with higher tensile strength bolts, with a reduction in seismic CDF to 1.1×10^{-5}/ry. In addition, the small LOCA median capacity was reduced from 1.5g (conservative) to 0.92g (based on NUREG/CR-

4840), with a calculated 4% increase in seismic CDF, indicating that overall seismic CDF is not very sensitive to the selection of small LOCA fragility parameters.

Only a few significant SDS sequences required additional non-seismic failures or human errors to result in core damage. These SDS sequences contribute less than 5% of overall seismic CDF, thus the base case seismic CDF result is not very sensitive to changes in modeling parameters. In addition, use of the EPRI mean hazard curve was found to result in a small (10%) reduction in seismic CDF.

In summary, the IP2 seismic IPEEE evaluates the CDF contribution for SDSs as opposed to determining accident sequence frequencies. Several sensitivity calculations and evaluations were performed to illuminate the impact of different assumptions on the CDF results.

2.1.10 Evaluation of Dominant Risk Contributors

Four SDSs were found to contribute about 92% of the seismic CDF. These SDSs are: (a) loss of instrumentation and control (due to structural failures of the turbine building frame and the Unit 1 superheater stack); (b) loss of CCW (due to failure of the CCW surge tank or failure of the steel superstructure of the fuel storage building); (c) loss of 480 V AC electric power (due to seismic failure of cable trays and the seismic failure of 480 V AC motor control centers [MCCs]); and (d) loss of service water (due to seismic failure of the pumps or heat exchangers, and sliding failure of the intake structure). A seismically-initiated LOSP together with non-seismic failures results in about 3% of the seismic CDF. Other SDSs contributing about 1% each are loss of the condensate storage and refueling water storage tanks (RWSTs), ATWS caused by failure of reactor internals, and ATWS with seismic failure of the RWST. None of these latter sequences contributes more than 5.0×10^{-7}/ry to seismic CDF, and none is considered to be a significant contributor to seismic CDF

In summary, the seismic IPEEE provides a meaningful description of seismic failures dominating CDF. Random failures and human errors have not been specifically identified; however, the submittal makes it clear that these non-seismic failures contribute less than 5% of the seismic CDF.

2.1.11 Relay Chatter Evaluation

In addition to performing an A-46 relay review, the licensee performed a low ruggedness ("bad actor") relay review for those relays that are associated with IPEEE-only equipment (i.e., equipment not on the A-46 SSEL). The review consisted of identifying IPEEE-only components, identifying the relays and contacts in their primary control and power circuits, evaluating the impact of chatter in secondary circuits, performing a bad-actor review by comparing the relays to the low capacity relay list in Appendix D of EPRI NP-7148-SL [14], and dispositioning any bad actors identified.

A total of 170 IPEEE-only components were identified. Following further review, only 83 components were found to employ relays in their control and/or actuation circuits. A total of 116 primary circuit relay coils were identified, none of which contained any bad-actor relay coils. In addition, 201 secondary circuit (interlock/actuation) contact pairs were identified in the primary circuits. Four of the contacts were found to originate from bad actor relays (all Westinghouse SC over-current relays used for protection of the station auxiliary transformer). Since chatter of these relays would at most result in a recoverable LOSP which would be lost for other reasons anyway during a significant seismic event, no remedial action was considered to be necessary.

In summary, the IP2 seismic IPEEE has involved an adequate bad-actor relay evaluation consistent with the guidelines of NUREG-1407. Although four bad actor relays were identified, these involved a recoverable LOSP which would be lost for other reasons during significant earthquakes.

2.1.12 Soil Failure Analysis

The potential for soil liquefaction and slope stability issues was addressed in the IP2 IPEEE by using the EPRI NP-6041, Rev. 1, procedures. IP2 is a rock site, and plant structures founded on rock were screened based on the EPRI guidance. Buried pipelines and tanks were specifically evaluated. While buried pipelines were found to be adequately protected, the diesel fuel oil tanks were found to be vulnerable to failure due to hold down strap failure and failure of grouted rock anchors. These failure modes were included in the seismic fragility evaluation.

A special review was performed of natural gas pipelines and "pig stations" located near IP2. As a conservative step, three potential failure impacts were evaluated: (a) a fire at the pipeline; (b) a potential explosion; and (c) transport of a vapor cloud and fire at the plant site. A fire at the pipeline was evaluated and determined not to impact IP2 because there is a 100-foot-wide firebreak around the plant. There is an old stack at the plant site which could collapse on the control room, but the IPEEE submittal indicates that natural gas does not detonate unless confined, and that therefore a severe shock wave at the plant site is not credible. Finally, the nearest point of approach of the pipeline is 1,200 feet from IP2. Natural gas is lighter than air and readily rises and disperses into the atmosphere. The IPEEE states that it is unlikely that weather conditions would form to support a gas cloud which could travel 1,200 feet and still support combustion or asphyxiation. A conservative bounding frequency calculation indicates that the frequency of an ignition of such a vapor cloud at IP2 is less than 6.0×10^{-7}/ry. The scenario was screened from further analysis based on this result and on the understanding that redundant and diverse systems would have to fail for the scenario to result in core damage.

In summary, the IPEEE submittal adequately addresses soil failure concerns. The conservative analysis of natural gas pipeline issues provides further support to the conclusion that no credible scenario exists.

2.1.13 Containment Performance Analysis

The IP2 IPEEE submittal follows the NUREG-1407 guidance concerning the seismic containment performance analysis. The major structures and systems whose failure could result in early containment failure or bypass were evaluated in the plant walkdowns and fragility evaluations. Containment penetrations were reviewed, and it was found that no isolation valves depend on air to provide closure capability. The containment personnel and equipment hatches were reviewed; no inflatable seals exist on the hatches. All sensors, transmitters, logic and relay cabinets, and power supplies for the containment isolation actuation system were included in the walkdowns and found to have high capacity. These components were therefore screened from the analysis. The seismic PRA results indicate that about 65% of the seismic CDF results in plant damage states with initial loss of containment pressure suppression and heat removal functions. If these functions are not regained, long-term overpressure failure of the containment could result. None of these sequences leads directly to early containment failure or bypass.

In summary, the containment performance assessment has examined all major failure modes, interaction issues, and other areas of concern. No vulnerabilities to early containment failure or bypass were noted in the assessment.

2.1.14 Seismic-Fire Interaction and Seismically Induced Flood Evaluations

The walkdowns performed as part of the IPEEE process were used to evaluate the potential for seismic-fire interactions. Sources of flammable gases and liquids were identified by systems engineers prior to the walkdown and from previous fire walkdown experience, and the seismic capacity of these flammable sources was evaluated by judgment. Seismic interactions which could affect these sources were also evaluated for seismic capacity.

Only two seismic-fire interactions could not be screened on the basis of the walkdown. The adequacy of the RCP lube oil collection tank seismic anchorage could not be confirmed. The tank provides standby capacity and does not normally contain large amounts of lube oil. Subsequent to the walkdown, it was determined that the anchorage of the tank was adequate. The second interaction not screened during the walkdown involved storage of hydrogen bottles near the alternate shutdown panel. Since alternate shutdown is not credited in the seismic analysis, this interaction was not considered to be significant.

The potential for seismically induced flooding and spray interactions was handled in an analogous manner. Special consideration was given to fire protection sprinkler heads in proximity to structural steel or other hard objects. Consideration of flooding from non-seismically designed tanks was based on input from the systems engineers and the internal flooding analysis. No potential interactions were identified as needing further consideration.

In summary, the seismic-fire and seismic-flooding interaction analysis in the IP2 IPEEE has addressed the major areas of concern.

2.1.15 Treatment of USI A-45

IP2 safety-related decay heat removal systems include the AFW system, the charging, SI, RHR, and recirculation systems, and the PORV system. Support systems include electric power, cooling water (service water and CCW), air/nitrogen, and room cooling and ventilation. Containment heat removal and pressure suppression are performed by the containment spray and fan cooler systems.

Each of these systems was included in the seismic PRA. All of the AFW equipment screened out of the seismic PRA except for the condensate storage tank (CST), which has a median fragility of 1.13g. Bleed-and-feed (use of PORVs to bleed primary coolant, and high pressure makeup to provide injection), a backup means of decay heat removal, was also evaluated. The front-line equipment associated with bleed-and-feed has high capacity except for the RWST, which has a median capacity of 0.61g. Sequences with seismic failure of the CST and combined with failure of the RWST or other bleed-and-feed failures, are estimated to contribute 3.0×10^{-7}/ry to seismic CDF (about 2% of the total).

Based on the relatively low CDF from earthquakes at IP2, conservatisms in the seismic modeling process, and the largest single contributor having a CDF of about 3.0×10^{-7}/ry, the licensee determined that no vulnerabilities in decay heat removal systems exist at IP2.

In summary, the seismic IPEEE has examined the capability of decay heat removal functions relevant to USI A-45. No vulnerabilities were identified. The evaluation approach addresses all of the major relevant seismic issues, and hence, the IPEEE's treatment of USI A-45 is judged to be appropriate and relevant to evaluation of decay heat removal concerns at IP2.

2.1.16 Treatment of GI-131

The IP-2 flux monitoring cart that is the subject of GI-131 has been modified to brace the cart in two directions. The flux monitoring cart was determined as part of the SPRA to have a HCLPF in excess of 0.5g. In accordance with the screening procedures used on the IP2 SPRA, the flux monitoring cart was screened from the analysis as seismically robust. On the basis of high seismic capacity, this issue may be considered to be resolved for IP2. As such, the IPEEE's treatment of GI-131 has adequately addressed the relevant concerns.

2.1.17 Other Safety Issues

The IP2 seismic IPEEE used both the EPRI and revised LLNL seismic hazard curves. These hazard curves directly address the Eastern U.S. Seismicity Issue. Thus, the IPEEE has followed the guidelines for resolving this issue.

With regard to USI A-46, the submittal notes several coordination efforts that improved the effectiveness of the IPEEE study. The submittal indicates that some of the same seismic experts were used for walkdowns in the A-46 and IPEEE studies; that system engineers provided systems expertise to the A-46 team and identified a list of "IPEEE-only" components; the USI A-46 team passed walkdown insights to the IPEEE program; and fragility calculations from A-46 were an important information resource for the IPEEE study. USI A-46 is being resolved separately from the seismic IPEEE. It should be noted that the scope of the A-46 program included the seismic spatial interaction aspects of USI A-17, as did the IPEEE seismic PRA walkdowns. In addition, it should be noted that the scope of USI A-46 included the USI A-40 concerns regarding the seismic capability of large safety-related above-ground tanks. The tanks (CST and RWST) were included in the SPRA as well, and no vulnerabilities were noted in this regard.

In summary, the Charleston Earthquake Issue, coordination of the IPEEE with USI A-46, the seismic spatial interactions aspect of USI A-17, and the seismic capacity of large above-ground tanks issue of US A-40 have all been appropriately treated in the seismic IPEEE.

2.1.18 Process to Identify, Eliminate, or Reduce Vulnerabilities

Although no formal definition for vulnerability was proposed in the IPEEE submittal, the submittal does reference the NEI severe accident closure guidelines [9]. The seismic IPEEE process has been thorough and generally well-executed in a search for vulnerabilities. None have been identified.

Overall, it may be stated that the IPEEE is capable of finding seismic-related severe accident vulnerabilities, but that none were identified as a result of a thorough review.

2.1.19 Peer Review Process

A formal project plan, task plans, and a quality assurance plan were incorporated into the project scope to help ensure the technical adequacy of the analysis and the validity of the work performed. Work products were reviewed at each stage by project team members, as well as by personnel other than those performing the task.

Two independent peer reviews were also incorporated into the licensee's IPEEE process. An independent review was performed by an in-house team which drew on experienced personnel (independent from the project team). A peer review by a team of industry experts was also performed. Seismic walkdowns were performed as part of the outside expert review to help further assure the validity of the analyses.

In conclusion, the IP2 seismic IPEEE has apparently been subjected to a meaningful peer review process.

2.2 Fire

A summary of the licensee's fire IPEEE process has been described in Section 1.2. In this section, the licensee's fire evaluation is described in detail, and discussion is provided regarding significant observations.

2.2.1 Overview and Relevance of the Fire IPEEE Process

a. Method Selected for Fire IPEEE

The IPEEE submittal includes extensive discussions regarding the methods and data used in conducting the fire analysis. The fire analysis was conducted using a PRA approach to examine all fire scenarios which were determined to be potentially risk-significant based on a progressive screening approach. The screening was performed using FIVE methodology. Each scenario was then treated as a separate initiating event and propagated through the plant model, which was modified to reflect the fire-induced equipment failures associated with the respective scenario. This allowed fire failures to be combined with random failures.

b. Key Assumptions Used in Performing Fire IPEEE

The submittal lacks explicit discussion of the key assumptions used in performing the fire analysis. However, the following assumptions have been identified during the review:

1. Reactor sub-criticality is assumed to be successful in all cases.

2. Fire propagation was assumed not to occur from low voltage enclosed cabinets with continuous conduit cable entry. However, a study to demonstrate the effect of propagation from high voltage (greater than 440V) cabinets was performed. The contribution to CDF was approximately 2×10^{-6}/ry.

3. Welding fires damaging fixed combustibles and cable junction box fires were eliminated from consideration because the walkdown showed that no significant amounts of exposed combustibles are near junction boxes, and open flame welding is prohibited in critical fire zones.

4. If the mean damage time fell within the range of drill response times, a 50% probability of successful manual suppression was assigned.

c. Status of Appendix R Modifications

The licensee states that all Appendix R modifications at IP2 have been completed and uses Appendix R definitions in the IPEEE fire analysis. For example, the definitions of fire areas and compartments are used per Appendix R guidelines. To define the safety component list, the IPE model has been used.

d. New or Existing PRA

The IPEEE is a new fire study employing both the FIVE and PRA methodologies.

2.2.2 Review of Plant Information and Walkdown

a. Walkdown Team Composition

Several plant walkdowns were performed for the IP2 fire analysis. The main objective of these walkdowns was to gather plant data which could not be readily derived from documented sources, in order to perform the screening and detailed analyses, as well as complete the evaluation of the FRSS issues. The licensee describes in some detail the qualifications of the personnel involved in the analysis, and the participation of the licensee's in-house personnel in the preparation of the fire IPEEE. The team was composed of experienced engineers and analysts.

b. Significant Walkdown Findings

The scope of the walkdowns includes a wide variety of considerations. The submittal does not indicate that the walkdown team discovered any fire vulnerabilities as a result of the plant walkdowns.

c. Significant Plant Features

IP2 has installed an Alternate Safe Shutdown System to mitigate the potential effects of fire, particularly those that disable the 480V buses. This system is able to take power from black-start gas turbines (1,2 or 3) which would normally supply power to Unit 1 auxiliaries. However, transfer switches have been installed which redirect this power to selected Unit 2 components. Instrumentation needed to operate the plant if the control room must be abandoned is distributed throughout the plant.

2.2.3 Fire-Induced Initiating Events

a. Were Initiating Events Other than Reactor Trip Considered?

Components and initiators modeled in the IPE have been used in the fire IPEEE. The analysis utilizes the IP2 IPE initiating event categorization, except for those initiating events which cannot be induced as a result of fire (e.g., steam generator tube rupture or steam line break). The categories of fire-induced initiating events which were evaluated include:

- General transient
- LOSP
- Small-break loss of coolant accident (SLOCA)

b. Were the Initiating Events Analyzed Properly?

The initiating events (IPE initiators) have been addressed in the IPEEE. The list of initiating events considered significant is typical of past PRAs for similarly configured plants.

2.2.4 Screening of Fire Zones

a. Was Proper Screening Methodology Employed?

Screening of fire zones appears to have been performed properly, albeit with the qualitative criterion found in the original FIVE study. Screening has been conducted in multiple stages, using the protocol prescribed as part of the FIVE methodology.

In the first qualitative stage, all safe shutdown equipment located in the compartment were considered failed, and the normal alternate shutdown equipment was considered unavailable. Given these conditions, if there is both a requirement for plant trip and the shutdown requires the use of equipment assumed to be damaged, then the area was retained for further analysis.

In the second stage, the CDF threshold of 1.0×10^{-6}/ry was applied. Using the fire occurrence frequency for each compartment, together with the CCDP, assuming loss of all cables and components in the compartment, the CDF contribution for the compartment was computed. Only one compartment was screened out in the second stage.

The submittal could have benefitted from more explanation with respect to definition of initiating events and justification for screening out the diesel generator building and RHR pump room.

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

Neither the cable spreading room nor the control room has been screened out of the analysis. The control room was subjected to a detailed analysis in which various panels were explored for potential fire occurrence, and the response of the operators to the fire event was considered.

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

No improperly screened fire areas/zones could be identified. This review finding was based on the information provided in the submittal. A random check of the frequencies and information on the type of equipment/cables present also revealed no errors or inconsistencies. The final results seem to be reasonable, and are within the expected range of results for PWR plants.

2.2.5 Fire Hazard Analysis

The FIVE methodology and data, along with a plant walkdown, have been used to estimate fire frequencies for individual compartments and fire areas. Weighting factors have been applied to apportion the overall fire frequency to the specific fire zones. Plant-specific fire occurrence data have not been used.

2.2.6 Fire Growth and Propagation

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

The submittal considered the potential for inter-area fire propagation from all compartments with fire loading greater than 20,000 BTU and into target compartments that could receive hot gases from the source compartment. After including a severity factor (the fraction of fires that are large enough to cause a significant hot gas layer) and assuming that suppression does not occur, the contribution to CDF was calculated to be approximately 7×10^{-6}/ry.

b. Assumptions Associated with Detection and Suppression

Automatic fire suppression failure rates were taken from FIVE. Specific consideration was given to detector and sprinkler spacing to ensure actuation prior to cable damage.

Manual fire suppression, prior to damage, was credited if the manual response during unannounced IP2 fire drills was shorter than the COMPBRN predicted time for cable damage. If the maximum drill time was less than the mean damage time, the success probability for manual suppression was assigned a value of 0.9. If the mean damage time fell within the range of the drill response times, the success probability for manual suppression was assumed to be 0.5.

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

The submittal reports that fixed fire suppression systems have not been installed where their operation or failure could cause unacceptable damage to safety-related equipment. A quantitative risk analysis performed as part of the GI 57 study [15] confirms the licensee's qualitative analysis.

d. Computer Code Used, if Applicable

The COMPBRN fire code [16] has been used in the fire propagation analysis.

2.2.7 Evaluation of Component Fragilities and Failure Modes

a. Definition of Fire-Induced Failures

A detailed analysis of fire induced hot shorts in control boxes was provided. This analysis applied to boxes that contain multi-wired control cables. As is typically the case, hot shorts in power supply lines were not considered credible.

b. Method Used to Determine Component Capacities

Component capacities were assigned for cables, sensitive electrical components, electric motors, and relays/switches. The following damage temperatures were employed:

- Cables: 623° K
- Sensitive electrical components: 339° K
- Electric motors: 339° K
- Relays, switches: 433° K

The damage threshold temperatures utilized by the licensee are consistent with those employed in past fire PRAs.

c. *Generic Fragilities*

As discussed above, fragilities have been subdivided for four classes of equipment. All fragilities were expressed in terms of a damage threshold temperature.

d. *Plant-Specific Fragilities*

No plant-specific fragilities have been mentioned in the submittal.

e. *Technique Used to Treat Operator Recovery Actions*

The following five categories of operator action were considered:

- Short-term control room actions (within the first four hours)
- Short-term local recovery actions
- Long-term operator actions in the control room modeled in the IPE
- Short-term local actions added to account for specific, post-fire recovery
- Long-term actions added to account for specific, post-fire recovery

Each short-term action was checked to ensure that the action is not prohibited by the presence of fire. For long-term operator recovery actions, the licensee assumed that modification of IPE recovery probabilities was unnecessary given that most actions did not require execution in less than 10 hours.

2.2.8 Fire Detection and Suppression

Fire detection and suppression were modeled explicitly for many fire scenarios. The combined time of detection and suppression was examined with respect to potential equipment and cable damage. Manual fire-fighting, and the effect of fixed fire suppression systems on the formation of a hot gas layer, have been considered.

2.2.9 Analysis of Plant Systems and Sequences

a. *Key Assumptions Including Success Criteria and Associated Bases*

The success criteria are taken directly from the IPE analysis and have not been modified for the fire analysis.

b. *Event Trees (Functional or Systemic)*

Each fire scenario was identified as an initiating event and the IPE event tree was requantified setting the probability of failure of the damaged equipment to 1.0.

c. *Dependency Matrix, if it is Different from Seismic Events*

No dependency matrix has been provided in the submittal.

d. *Plant-Unique System Dependencies*

The submittal does not present any unique system dependencies.

e. Most Significant Human Actions

Human actions, as discussed above, have been considered as an integral part of the fire scenario quantification. The submittal does not summarize the most significant human actions. The Human Cognitive Reliability (HCR) model was used to evaluate operator actions and post-fire recovery actions.

2.2.10 Fire Scenarios and Core Damage Frequency Evaluation

Using a PRA methodology, several fire scenarios have been identified and analyzed. The analyses have taken into account numerous relevant issues, including the list of equipment and components failed by a fire, the CCDP given damage caused by the fire, and the conditional probability of failure to suppress the fire. A large number of fire areas and compartments have been considered, and a summary of the analysis is presented in the submittal. The control room was analyzed in detail—the possibility of operators abandoning the control room was explicitly considered and modeled.

CDF was the principal parameter used for quantitative screening in the fire analysis. Plant damage models (i.e., IPE models) have been used in the analysis. Fire ignition, propagation, detection, and damage have been quantified. The IPEEE submittal provides sufficient detail to allow verification of a chain of computations, and presents numerous tables and figures that assist in this regard.

2.2.11 Analysis of Containment Performance

a. Significant Containment Performance Insights

Containment performance was evaluated for the potential of fire-induced containment bypass and failure of containment isolation. Mechanical or spurious valve operation due to control and power circuit damage was evaluated, and the licensee found no significant fire-induced bypass or failure of isolation mechanisms. The fire analysis did not evaluate containment pressure suppression and heat removal system status.

b. Plant-Unique Phenomenology Considered

No containment-related event trees have been used in any of the screening phases, nor in evaluating the unscreened fire zones.

2.2.12 Treatment of Fire Risk Scoping Study Issues

a. Assumptions Used to Address Fire Risk Scoping Study Issues

All of the Sandia FRSS issues have been addressed closely following the methodology described in FIVE. The licensee has presented a discussion pertaining to each issue, as summarized below.

1. Seismic-fire interaction has been addressed by examinations of the potential for: (a) a fire event resulting from an earthquake; (b) inadvertent seismic actuation of the fire suppression systems and resulting effects on safety equipment; and (c) seismically induced failure of the fire protection

system. A walkdown of the plant has been undertaken for these examinations. The licensee did not identify any potential vulnerabilities.

2. The submittal concludes that the potential for fire barrier failure is not risk-significant, based upon periodic surveillance programs.

3. In the assessment of manual suppression effectiveness, the submittal states that a program is in place to indoctrinate selected plant personnel in the administrative procedures that implement the IP2 fire protection program. Orientation of plant personnel in the use of fire extinguishers is provided by general employee training. Also, the fire brigade conforms with the Appendix R requirements.

4. The licensee concludes that the potential effects of non-thermal products of combustion on safety equipment are insignificant. Regarding operator effectiveness, self-contained breathing apparatus (SCBA) equipment is provided, as well as emergency lighting units.

5. An assessment of IP2 post-fire alternative shutdown features was performed to determine the design/operational characteristics applied to the normal and alternative equipment trains. Particular attention was given to any alternative trains that rely on a control transfer and/or shared equipment scheme. The licensee concluded that the design of the IP2 alternative shutdown capabilities is "generally immune" to the effects of control system interactions.

b. Significant Findings

The following are the significant findings associated with the FRSS issues:

1. The suppression systems do not have a significant potential to adversely affect safety systems.

2. Procedures are available that address fire-related issues.

3. Existence of an alternative shutdown capability, and the provision for isolating this capability, minimize the potential for control systems interaction.

2.2.13 USI A-45 Issue

a. Methods of Removing Decay Heat

The IPEEE fire analysis uses logic models developed for the IPE, which include the entire array of heat removal capabilities of the plant.

b. Ability of the Plant to Feed and Bleed

The IPE model used in the IPEEE includes the provision for feed and bleed cooling.

c. Credit Taken for Feed and Bleed

Credit has been taken for feed and bleed capability.

d. Presence of Thermo-Lag

The submittal provides no information concerning the presence of Thermo-lag.

2.3 HFO Events

Based on an analogy to NEI guidance with respect to severe accident closure, the IP2 IPEEE submittal reports no vulnerabilities with respect to HFO events. The submittal indicates that implementation of the screening approach described in Supplement 4 to GL 88-20, and in the guidance of NUREG-1407, has formed the basis for the conclusion that the plant is not vulnerable to HFO events.

The general methodology that has been implemented for the HFO events analysis makes use of the following screening steps:

1. High winds, external floods, and transportation and nearby facility accidents have been considered using the approach described in NUREG-1407. Other plant external hazards have been considered using the screening methods described in the PRA Procedures Guide [17].

2. For high winds, external floods, and transportation and nearby facility accidents, any changes that have taken place since the time the operating license (OL) was issued have been identified.

3. For high winds, external floods, and transportation and nearby facility accidents, a quick screening has been performed to identify whether or not the plant meets the 1975 SRP criteria. As a result, further analysis was performed for winds, floods and some of the transportation and nearby facility accidents.

4. The resulting HFO events analysis has been documented in the IPEEE submittal report.

The licensee conducted a walkdown with the objective of collecting information on HFO events. Concurrent with the walkdown activities, a review was made of plant design documents, including the Updated Final Safety Analysis Report (UFSAR), the IPPSS, and recent meteorological data collected by the licensee. The walkdown and survey of the area within 5 miles of the plant was performed to confirm that no significant changes to the plant, and in the site region, have occurred since the issuance of the OL and the IPPSS. The walkdown concentrated on outdoor facilities that could be affected by the external events addressed in this section (with emphasis on high winds and onsite storage of hazardous materials), and on offsite developments. The walkdown was performed following procedures developed specifically for the IPEEE, and included engineering and technical personnel from both the utility (four representatives) and the contractor team (three representatives).

Due to the importance of high winds identified in the original IPPSS, the onsite walkdown concentrated on outdoor tanks and equipment; entrances to concrete buildings; openings in buildings such as air intakes, diesel exhaust stacks, and louvers; block walls in structures with openings; structures which could collapse and impact buildings containing safety-related equipment; and availability of objects which could become missiles in a tornado or hurricane. The main purpose of the walkdown was to obtain an overall appreciation of the plant layout, location of structures, and the types of construction, and generally confirm the validity of structural drawings for tanks and buildings from which most of the information for wind fragility evaluation was obtained. The walkdown activities also included inspection of the area surrounding the immediate site, and contact with cognizant non-utility personnel with knowledge of current conditions and activities which could impact the examination.

2.3.1 High Winds and Tornadoes

2.3.1.1 General Methodology

IP2 structures and systems were designed to the wind loading requirements of the building codes in effect in the early 1970s. They predate, and do not meet, the 1975 SRP criteria. Also, some of the structures at IP2 housing safety-related equipment are metal-sided steel structures offering limited resistance to tornado missiles. The extreme wind hazard analysis done in the IPPSS indicated that high winds could not be screened out on the basis of low frequencies of occurrence. Therefore, utilizing the NUREG-1407 screening approach, the licensee concluded that a detailed PRA was needed to address the impact of high wind events at IP2.

2.3.1.2 Plant-Specific Hazard Data and Licensing Basis

The wind hazard and building fragility analysis performed in the IPPSS analysis was reviewed and updated. An event-tree-based approach was used to define a set of unique wind-induced plant damage states.

In the IPPSS, simplified fault tree models were developed to represent the various combinations of wind-initiated events (including hurricanes, extratropical cyclones, tornadoes and tornado missiles) and resulting equipment failures which may lead to core damage. Three types of scenarios were initially considered: transients coupled with failure of decay heat removal, loss of RCP seal cooling (resulting in seal LOCA) coupled with failure of SI, and large LOCA coupled with failure of SI. Wind-induced large LOCAs were subsequently ruled out.

Data on occurrence of tornadoes, hurricanes, extratropical cyclones, and thunderstorms were taken from the IPPSS. As a check, the IPEEE analysis reviewed the tornado occurrence rate data. Since, based upon a preliminary quantification by the licensee, tornado-induced mean frequency of core damage was significantly greater than that induced by hurricanes, qualitative review of the IPPSS hurricane hazard analysis was deemed sufficient.

2.3.1.3 Significant Changes Since Issuance of the Operating License

The submittal states that a walkdown was performed to ensure that no significant changes have occurred since the time of issuance of the OL.

2.3.1.4 Significant Findings and Plant-Unique Features

The submittal states that equipment located inside concrete buildings (i.e., reactor building and lower portion of the primary auxiliary building and auxiliary feed pump building) are generally protected from wind loading and missile penetration. Equipment located within sheet metal clad structures are partially protected (i.e., top portions of the primary auxiliary building, AFW structure, turbine generator building, and gas turbine generator building). Equipment in the yard (e.g., CST, service water pump) are not protected from tornado- or hurricane-induced missiles.

2.3.1.5 Hazard Frequency

The IPEEE analysis separates the effects of hurricanes from those of tornadoes and extratropical cyclones. At each wind speed, the wind speed exceedance probabilities for tornado and extratropical cyclones were added to obtain wind hazard curves for the combined extratropical cyclone and tornado event. The submittal presents median capacities for the key structures at IP2. The median capacities vary from 83 mph (e.g., gas turbine shelters) to 222 mph (e.g., auxiliary feed pump building). For wind speeds of 100 mph or less, the wind speed exceedance frequency is dominated by hurricanes (4.0×10^{-6}/yr and greater), while for wind speeds of 125 mph or greater, the wind speed exceedance probability is dominated by tornadoes (3.0×10^{-6}/yr and less).

2.3.1.6 Bounding Analysis

The IPEEE study has used a PRA analysis to further analyze tornado, extratropical cyclone, and hurricane events. Therefore, bounding analyses were not performed.

2.3.1.7 PRA Analysis

The PRA analysis determined the CDFs resulting from each wind damage state, taking into account equipment loss due to wind-related structural damage and tornado missile damage, as well as unrelated coincident-random equipment failures. Within the internal events model, the general transient event tree was selected by the licensee for the purpose of modeling accident sequences resulting from wind-induced initiating events. The model included the potential for, and mitigation of, consequential LOCA events resulting from loss of RCP seal cooling or a stuck-open PORV. LOSP was reflected in the support systems logic, as was the recovery of power from the gas turbines.

The total contribution to CDF from all three wind hazard types is 3.0×10^{-5}/ry. The containment analysis found that 87% of wind CDF leads to station blackout, loss of all containment heat removal, and long term containment failure (if no recovery). The contribution to CDF from each wind damage state and wind hazard type is given in the submittal. Tornadoes and extratropical cyclones are the major HFO contributors to CDF, contributing 1.7×10^{-5}/ry and 1.1×10^{-5}/ry, respectively. Hurricane events contribute 2.4×10^{-6}/ry.

The dominant sequences for tornado and extratropical cyclones occur due to wind damage state "w02." In the case of tornadoes, failure of the turbine building (leading to consequential failure of the control building), the control building itself, and the combination of emergency diesel generator/gas turbine (EDG/GT) building failure all contribute significantly. In the case of extratropical cyclones, the EDG building failures are more important. Due to the resulting station blackout, RCP seal cooling is lost, resulting in a seal LOCA with no reactor coolant system (RCS) make-up capability.

There is also some contribution to core damage from scenarios which include missile damage (principally on the control and EDG buildings) and/or coincident random equipment failures (principally the gas turbines). In the case of tornadoes, this contribution is 3.8×10^{-6}/ry, whereas the contribution from extratropical cyclones is 3.5×10^{-6}/ry.

The contribution to CDF from hurricane events (2.4×10^{-6}/ry) is less significant compared with the other two wind hazards. This is the case for two reasons. First, and of greater significance, the frequencies associated with hurricane wind speeds in the range which could cause severe plant damage are substantially lower than those associated with tornadoes and extratropical cyclones. Second, and of lesser significance, the implementation of the IP2 hurricane technical specification and implementing procedure requires the plant to be in a cold shutdown condition prior to the hurricane reaching the site. In this condition, the likelihood of an RCP seal LOCA is reduced, and the plant may be maintained in a stable condition using the turbine-driven AFW pump and pneumatic instrumentation.

2.3.2 External Flooding

2.3.2.1 General Methodology

The grade elevation at the plant embankment adjoining the river is 14.0 feet, and rises above this level at all other plant buildings and structures. The minimum critical flood height for IP2 is in the 480 V Switchgear Room, at elevation 15.5 feet. The probable maximum flood (PMF) analysis conducted in 1971-73, during licensing of the adjoining IP3 facility, concluded that the maximum sustained water surface elevation at the plant is 14.0 feet, based on the combination of a Hudson River maximum flood, probable maximum precipitation over the Esopus Creek Basin resulting in failure of the Ashokan Dam, and a hurricane at New York Bay.

Probable maximum precipitation (PMP) was evaluated for six storm events, one of which was a 100-year, 24-hour storm event. The PMP analysis provides the locations and maximum depths of ponds which form on the roofs of various site structures and in various areas around the site. The roofs were evaluated by converting the water depth to a loading, and comparing the calculated loading with the allowable roof loadings.

2.3.2.2 Plant-Specific Hazard Data and Licensing Basis

The plant has not experienced flooding from the Hudson River that has exceeded the plant grade elevation. Information obtained from the U.S. Army Corps of Engineers by the licensee has confirmed that there is a stream gage on the Hudson River (at Greenland), and that the water level does not exceed 14.0 feet above mean sea level (MSL). Since the river is very wide, its water depth does not fluctuate much. The walkdown and review of the surrounding site showed that no major construction has taken place that may change the river regime upstream of the Hudson River since the IPPSS, nor has there been any major changes to the terrain around the plant. Therefore, the licensee concluded that the response of the terrain to a hazard which could cause river flooding, as evaluated in the IPPSS, is still valid.

With regard to hurricane-induced river flooding, more recent hurricane inundation maps for Westchester County developed by the U.S. Army Corps of Engineers and the Federal Emergency Management Agency show that the maximum hurricane surge elevation for a Category 4 hurricane, were it to occur close to Indian Point, could reach 13.5 feet. This maximum surge could only occur for the Category 4 hurricanes with wind speeds at the upper end of the range for the category.

2.3.2.3 Significant Changes Since Issuance of the Operating License

The submittal states that a walkdown was performed to ensure that no significant changes have occurred since the time of issuance of the OL.

2.3.2.4 Significant Findings and Plant-Unique Features

No significant findings related to flood events were reported.

2.3.2.5 Hazard Frequency

External flooding was screened out due to the elevation of IP2. The IPPSS study estimated the annual frequency of the combination of extreme events leading to the PMF to be in the range of 1.0×10^{-8} to 1.0×10^{-12}/ry. Therefore, the licensee concluded that the contribution of external flooding to CDF at IP2 is extremely small.

In consideration of the PMP, the buildings which contain safety-related equipment and which were also considered to be susceptible to ponding are the primary auxiliary building, AFW building, turbine building, and control building. Using the maximum allowable live loading for these buildings, an equivalent maximum allowable height of water accumulation was calculated and compared against the maximum height of accumulated rainfall on those buildings. Only the turbine building could experience loads at, or close to, yield. However, given the conservatisms in the hazard calculation and the remaining margin between yield and actual failure stress, the licensee judged that the structure would remain intact.

2.3.3 Transportation and Nearby Facility Accidents

2.3.3.1 Methodology

The IP2 IPEEE submittal has addressed aircraft crashes, as well as water, rail, and highway transportation accidents. Also, the submittal considers potential impacts of on-site hazardous material inventories.

Airports and airfields within approximately 25 miles of Indian Point were considered in the IPPSS study. The three closest airports were identified as Mahopac, Ramapo Valley and Peekskill Seaplane Base, out of which the Peekskill Seaplane Base was judged to pose the greatest hazard to the plant. Using the annual number of landing and take-off operations at the Seaplane Base, and general aviation accident statistics, the annual probability of an aircraft hitting any of the plant structures was estimated as 2.4×10^{-7}/ry. Federal airways in the vicinity of the plant were also examined. The annual frequency of an aircraft using the federal airways in the vicinity of the plant and accidentally hitting IP2 structures was estimated by the licensee to be 4.6×10^{-8}/ry.

The nearest rail facilities are located approximately 0.9 miles west and 0.6 miles east of the plant site. The closest distance to the rail lines from the plant is larger than the stand-off distance. Therefore, the licensee concluded that IP2 meets the 1975 SRP requirements for rail transportation.

The nearest major road is New York Highway 9 extending north/south and located between one to two miles east of the plant site. The distance to the road is much larger than the stand-off distance. Therefore, the licensee concluded that IP2 meets the 1975 SRP requirements for road transportation.

The potential consequences of accidents involving barges on the Hudson River are overpressure on the structures due to explosion, fire at the shoreline, and release of toxic chemicals. The annual frequency of a large, rapid spill resulting in a fire at the shoreline was estimated by the licensee to range from 1.0×10^{-6}/ry to 1.0×10^{-9}/ry. With respect to potential damage due to detonation of explosive gases, IP2 is located on the shore of the Hudson River and therefore cannot be screened using the safe stand-off distance criteria. The frequency of barge accidents resulting in overpressures exceeding 1 psi was determined to be 3.9×10^{-6}/ry.

There are two underground natural gas transmission lines (26-inch and 30-inch diameter) passing through the IP2 site about 1,000 feet from the closest plant structures. The frequency of failure of these pipelines which could pose a hazard to the plant was stated in the submittal to be about 5.0×10^{-7}/ry.

A number of toxic chemicals stored at IP2 were identified by the licensee. The major potential hazardous chemical emission sources are a 10-ton CO_2 cylinder at IP3, and a 1-ton chlorine cylinder at Peekskill Sewage Disposal Plant. The control room ventilation intake has quick response chlorine detectors. Using "worst case" chemical release conditions, the estimated maximum control room gas concentrations were determined to be less than 1% CO_2, and less than 1 ppm chlorine.

2.3.3.2 Plant-Specific Hazard Data and Licensing Basis

For aircraft crashes, the IPPSS study was used to determine the frequency of a crash. The submittal is organized and systematic in the analysis. Some data on the frequency of flights from nearby airports is reported.

For the analysis of land and water transportation events, most were screened based on stand-off distance criteria. In the case of barge traffic, the licensee states that the conditional probability of core damage, given a 1-psi overpressure, is 0.1. Therefore, barge accidents were eliminated from further consideration.

Toxic chemical accidents have been evaluated in detail using "worst case" meteorological conditions. Toxic gas concentrations in the control room (even without ventilation system isolation) were stated to be less than Emergency Response Planning Guidelines-2 (ERPG-2) concentrations.

Natural gas pipeline accidents were screened based upon the frequency of such accidents which could pose a hazard to the plant.

2.3.3.3 Significant Changes Since Issuance of the Operating License

The submittal states that there have not been any significant developments that affect the original design condition with regard to transportation and nearby facility accidents since the issuance of the OL.

2.3.3.4 Significant Findings and Plant-Unique Features

All these hazards screen out.

2.3.3.5 Hazard Frequency

For aircraft crash, barge overpressure, and pipeline accidents, hazard frequency arguments have been utilized to screen these events from further consideration. Some details of the hazard frequency determination are provided. Based upon separation distance, the aircraft crash and natural gas pipeline accident analyses are considered reasonable. In the case of barge accidents, sufficient detail is provided by the submittal to support the conclusion that this hazard screens.

2.3.4 Other HFO Events

2.3.4.1 General Methodology

The screening process followed in NUREG-1407 was reviewed by the licensee in light of IP2-specific information. In addition, any known external hazards that may have the potential to damage IP2 were examined. The external hazards and the screening criteria listed in Reference [17] were used. A standard table describing the rationale for screening of each "other" external hazard is provided in the submittal. The licensee concluded that there were no other plant-unique external events which pose a significant hazard to the plant.

2.3.4.2 Plant-Specific Hazard Data and Licensing Basis

The IPPSS study was utilized to screen most "other" external events. Also, if the external event could, at most, result in a LOSP, it was eliminated from further consideration.

2.3.4.3 Significant Changes Since Issuance of the Operating License

The submittal states that a walkdown was performed to ensure that no significant changes have occurred since the time of issuance of the operating license.

2.3.4.4 Significant Findings and Plant-Unique Features

No significant findings are discussed in the submittal for "other" HFO events.

2.3.4.5 Hazard Frequency

In a few cases, hazard frequencies derived in the IPPSS study are utilized to screen some "other" HFO events. For most "other" HFO events, PRA Procedure Guide criteria are employed for screening.

2.4 <u>Additional Generic Issues (GSI-147, GSI-148, GSI-156, GSI-172)</u>

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

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

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

The submittal has followed the guidance provided in FIVE concerning control system interactions. A detailed assessment was performed by the licensee to determine the design/operational characteristics applied to the normal and alternative trains. Particular attention was given to any alternative trains that rely on a control transfer and/or shared equipment scheme.

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
- Electronic equipment can be damaged or degraded
- By hampering the operator's ability to safely shutdown the plant
- By initiating automatic fire protection systems in areas away from the fire

Reference [18] identifies possible reduction of manual fire-fighting effectiveness and causing misdirected suppression efforts as the central issue in GSI-148. Manual fire-fighting was credited in the analysis. The hindering of short-term (less than four hours) operator recovery actions due to smoke was considered, and IPE recovery probabilities were modified as appropriate. No discussion is provided in the submittal regarding either smoke-induced misdirection of manual suppression efforts or actuation of automatic FPSs in areas away from the fire.

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

GSI-156 addresses issues encountered at plants that were licensed prior to the time the 1975 Standard Review Plan (SRP) was issued. Among other concerns, GSI-156 issues relate to seismic; fire; and high winds, floods, and other (HFO) external events. Reference [18] 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 [18]: 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.

Static settlements are not believed to be a concern, and 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.

Section 3.1.3.5 of the Indian Point Unit 2 IPEEE submittal provides a discussion of this issue. IP2 is a rock site and as such does not have associated with it any soil failure issues. The plant was screened for such issues during the seismic capability walkdowns. Buried pipelines inside the plant fence are run in trenches excavated from rock and backfilled. Some settlement could occur, but the amount of backfill beneath the pipe is expected to be small, and any seismically-induced settlement would not be of sufficient magnitude to fail the buried pipes. The only important buried tanks are the diesel fuel oil tanks, and these tanks were not considered to be of concern for the same reasons as the buried piping.

Dam Integrity and Site Flooding

Description of the Issue [18]: 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 1982-1983 IPPSS study evaluated the frequency of PMP and failure of an upstream dam leading to flooding at the plant site to be less than 1.0×10^{-8}/ry (Section 6.3 of the submittal). There is no discussion in the submittal, however, of seismically-induced dam failure.

Site Hydrology and Ability to Withstand Floods

Description of the Issue [18]: 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 Indian Point IPEEE submittal includes a discussion of external floods, including the effects of storm surge and probable maximum precipitation, in Sections 6.3 and 6.6.

Industrial Hazards

Description of the Issue [18]: 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 Indian Point IPEEE submittal (Section 6.4) includes the following information of relevance to this issue: Section 6.4.1 discusses aircraft accidents; Section 6.4.2 discusses other transportation accidents; Section 6.4.3 discusses gas pipeline accidents; and Section 6.4.4 discusses release of toxic chemicals.

Tornado Missiles

Description of the Issue [18]: 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 Indian Point IPEEE has involved an evaluation of tornadoes, including tornado-induced missiles, and a detailed discussion is provided in Section 6.2 of the submittal.

Severe Weather Effects on Structures

Description of the Issue [18]: 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 Indian Point IPEEE has included evaluations of high winds (straight wind loads, hurricanes, and tornadoes) and external floods. Submittal Section 6.2 discusses high winds, and Section 6.3 discusses external flooding.

Design Codes, Criteria, and Load Combinations

Description of the Issue [18]: 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).

No discussion of design codes, criteria, and load combinations is provided in the IP2 IPEEE submittal.

Seismic Design of Structures, Systems, and Components

Description of the Issue [18]: 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 IP2 seismic IPEEE includes an evaluation of seismic capability of plant structures and equipment. See Section 3.1.4 of the submittal for details.

Shutdown Systems and Electrical Instrumentation and Control Features

Description of the Issue [18]: 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.

This issue was addressed as part of the IP2 IPEEE, and pertinent information is provided in Sections 3.2.1 and 4.9 of the submittal.

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

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

Common Cause Failures Related to Human Errors

Description of the Issue [18]: Common cause failures (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.

No discussion on the potential for human-error-caused common cause failures (CCFs) unique to external events is highlighted in the IP2 IPEEE submittal. Human errors were included in the PRA models, but were limited to errors of omission (which is consistent with typical PRA practice). Human errors (together with random hardware failures) were found to be insignificant contributors to the seismic CDF (less than 5%). For fire events, treatment of human errors is discussed in Section 4.6.1.2 of the IPEEE submittal.

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

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

Control system dependencies were treated directly in the PRA models for seismic and fire events. No control system dependencies were identified as contributing to risk for HFO events. Control system dependencies were also addressed in the USI A-46 program. Information provided in the IP2 IPEEE submittal pertaining to seismically induced spatial and functional interactions is identified below (under the heading *Seismically Induced Spatial and Functional Interactions*), whereas information pertaining to fire-induced alternate shutdown and control panel interactions has already been identified in Section 2.4.1 of this TER.

Heat/Smoke/Water Propagation Effects from Fires

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

- Heat, smoke, and water may propagate (e.g., through heating, ventilation and air conditioning (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 Indian Point 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.

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

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

This issue was addressed in the seismic capability walkdowns (Section 3.1.3 of the IPEEE submittal). In addition, Section 4.8.5 of the IPEEE submittal states that fire suppression systems have not been installed where operation or failure could cause unacceptable damage to safety-related equipment. A quantitative risk analysis was performed as part of the GSI-57 study [15], which confirms the licensee's qualitative analysis.

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

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

The Indian Point IPEEE submittal discusses external flooding in Section 6.3.

Seismically Induced Spatial and Functional Interactions

Description of the Issue [18]: Seismic events have the potential to cause multiple failures of safety-related systems through spatial and functional interactions. Some particular sources of concern include: ruptures in small piping that may disable essential plant shutdown systems; direct impact of non-seismically qualified structures, systems, and components that may cause small piping failures; seismic functional interactions of control and safety-related protection systems via multiple non-safety-related control systems' failures; and indirect impacts, such as dust generation, disabling essential plant shutdown systems. As part of the IPEEE, it was specifically requested that seismically induced spatial interactions be addressed during plant walkdowns. The guidance for performing such walkdowns can be found in EPRI NP-6041.

Seismically induced spatial and functional interactions were addressed in the seismic capability walkdowns performed as part of the IP2 seismic IPEEE. The walkdowns are discussed in Section 3.1.3 of the submittal.

Seismically Induced Fires

Description of the Issue [18]: Seismically induced fires may cause multiple failures of safety-related systems. The occurrence of a seismic event could create fires in multiple locations, simultaneously degrade fire suppression capability, and prevent mitigation of fire damage to multiple safety-related systems. Seismically induced fires is one aspect of seismic-fire interaction concerns, which is addressed as part of the Fire Risk

Scoping Study (FRSS) issues. (IPEEE guidance specifically requested licensees to evaluate FRSS issues.) In IPEEEs, seismically induced fires should be addressed by means of a focused seismic-fire interactions walkdown that follows the guidance of EPRI NP-6041.
Seismically induced fires were addressed in the seismic capability walkdowns performed as part of the IP2 seismic IPEEE. The walkdowns are discussed in Section 3.1.3 of the submittal. Seismically induced fires were also discussed as part of the Fire Risk Scoping Study issues in Section 4.8.2 of the IPEEE submittal.

Seismically Induced Fire Suppression System Actuation

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

Seismically induced fire suppression system actuation was addressed in the seismic capability walkdowns performed as part of the IP2 seismic IPEEE. The walkdowns are discussed in Section 3.1.3 of the submittal. Seismically induced fire suppression system actuation was also addressed as part of the Fire Risk Scoping Study Issues in Section 4.8.2 of the IPEEE submittal.

Seismically Induced Flooding

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

Seismically induced flooding was addressed in the seismic capability walkdowns performed as part of the IP2 seismic IPEEE. The walkdowns are discussed in Section 3.1.3 of the submittal.

Seismically Induced Relay Chatter

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

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

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

Seismically induced relay chatter was addressed in Section 3.3 of the IP2 IPEEE submittal.

Evaluation of Earthquake Magnitudes Greater than the Safe Shutdown Earthquake

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

The IP2 IPEEE documents the performance of a seismic probabilistic safety assessment (PSA). The seismic input for the PSA is provided and discussed in Section 3.1.1 of the submittal. Section 3.1.6 of the submittal includes a table summarizing the contribution to seismic CDF of various acceleration ranges, indicating that earthquakes with horizontal PGA from 0.05g-0.25g contribute only 6% of the seismic CDF.

Effects of Hydrogen Line Ruptures

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

The effects of earthquakes on gas lines was addressed in the seismic capability walkdowns performed as part of the IP2 seismic IPEEE. The walkdowns are discussed in Section 3.1.3 of the submittal. Hydrogen fire sources are discussed in the fire portion of the submittal in Section 4.3.2.2.

3 OVERALL EVALUATION AND CONCLUSIONS

3.1 Seismic

Judged on the basis of the submittal, this review concludes that the licensee's seismic IPEEE methodology is capable of identifying severe accident vulnerabilities. It appears that the licensee understands the plant and seismic PRA techniques, and has conscientiously applied this knowledge to produce the seismic IPEEE submittal. The IP2 seismic IPEEE is comprehensive with respect to the important points of GL 88-20 and NUREG-1407. Based on this submittal-only review, the following strengths of the IPEEE submittal for IP2 were identified (no significant weaknesses were identified):

Strengths

(1) The licensee demonstrated a good grasp of the application of seismic PRA technology.

(2) The equipment list was comprehensive.

(3) The relay chatter study was thorough and well coordinated with USI A-46.

(4) The plant familiarity and walkdown processes were well-structured and well-coordinated with USI A-46.

Weaknesses

None.

3.2 Fire

For the evaluation of fire initiators, the licensee demonstrated detailed knowledge of the plant and fire PRA methodology, and has made a conscientious application of this knowledge. The licensee has employed proper methodology (i.e., the EPRI FIVE methodology for screening, and a PRA methodology for CDF quantification), and has employed proper data bases and calculational methods for fire occurrence and suppression system failure rates. No scenario contributes more than 10% of the total CDF. The many low contribution scenarios are typical of the ignition source driven PRA method used. Notable strengths of the submittal include the following (no significant weaknesses were identified):

Strengths

(1) Assumptions, sensitivity studies and uncertainties are well presented.

(2) The inter-compartment fire propagation analysis was unusually thorough.

(3) The hot short analysis was far more comprehensive than is typical for an IPEEE.

The final conclusions of the submittal are reasonable, and are within the range of results expected for a pressurized water reactor (PWR). The licensee's fire IPEEE process is capable of identifying severe accident vulnerabilities and none were found.

Weaknesses

None.

3.3 HFO Events

The licensee's HFO evaluation implemented the progressive screening method of NUREG-1407. Because IP2 generally does not meet the SRP, additional analyses were performed, as needed. Hazard screening and verification walkdowns were done appropriately, and changes since issuance of the OL were noted. Good use was made of earlier PRA work. The analysis was comprehensive per NUREG-1407. No significant weaknesses were noted during this review. Noteworthy strengths are as follows:

Strengths

(1) A state-of-the-art wind PRA was performed.

(2) A good bounding analysis demonstrated that the PMF was less than grade.

Weaknesses

None.

4 IPEEE INSIGHTS, IMPROVEMENTS, AND COMMITMENTS

4.1 Seismic

The IP2 plant seismic mean CDF was estimated at 1.46×10^{-5}/ry (for the LLNL hazard input). The dominant seismic failure modes involved structural failures (turbine building or superheater stack) which are assumed to cause failure of plant instrumentation and control systems. Non-seismic (random) failures and human actions were included in the analysis (indeed, human error rates were increased to attempt to account for seismic effects on human reliability), but were not significant contributors to the CDF. The containment performance analysis identified no unique seismic failure modes. The low ruggedness relay evaluation demonstrated that potential relay chatter does not have adverse effects on plant equipment. Although the study resulted in the conclusion that there are no seismic vulnerabilities, the licensee nevertheless increased the strength of the CCW surge tank anchor bolts, which was one of the dominant risk contributors. Following this upgrade, the mean CDF was calculated to be 1.1×10^{-5}/ry.

4.2 Fire

The total fire CDF from "unscreened" scenarios is estimated at 1.8×10^{-5}/ry. This frequency is within the range of fire-induced CDFs obtained for other nuclear power plants (NPPs). The dominant contributors to the fire CDF are the control room, cable spreading room, and a switchgear room, though each contributes 10% or less to the total CDF.

The licensee has concluded that there are no significant fire vulnerabilities at IP2. The licensee has used NEI's severe accident closure guidelines [9] to evaluate the need for plant improvements. No improvements or commitments were identified as being necessary to further reduce the fire risk at IP2.

4.3 HFO Events

The licensee has concluded that there are no significant HFO vulnerabilities at IP2. No improvements or commitments were identified as being necessary to further reduce the HFO risk at IP2. The licensee has used NEI's severe accident closure guidelines to evaluate the need for plant improvements.

5 IPEEE EVALUATION AND DATA SUMMARY SHEETS

Completed IPEEE evaluations and data summary sheets for the IP2 IPEEE are provided in Tables 5.1 to 5.8. These tables have been completed in accordance with the descriptions in Reference [11]. Table 5.1 lists the overall external events results, and Table 5.2 summarizes the important seismic fragility values. Tables 5.3 to 5.5 provide PWR Accident Sequence Overview Tables for seismic, fire, and high winds events, respectively. Tables 5.6 to 5.8 provide the PWR Accident Sequence Detailed Tables for seismic, fire, and high winds events, respectively. Note, for Tables 5.3 to 5.8, the submittal does not provide sufficient detail to fully complete these tables.

Table 5.1
External Events Results

Plant Name: Indian Point Unit 2

Event	Screening	CDF	Plant HCLPF(g)	Notes
External Flooding	O			
Extreme Winds	S	Tornadoes: $1.7 \cdot 10^{-5}$/ry Extratropical Cyclone: $1.1 \cdot 10^{-5}$/ry Hurricane: $2.4 \cdot 10^{-6}$/ry		
Internal Fire	S	1.8×10^{-5}/ry		
Nearby Facility Accidents	O			
Seismic Activity	S	Before CCW Fix: 1.46×10^{-5}/ry After CCW Fix: 1.1×10^{-5}/ry		LLNL hazard input
Transportation Accidents	O			
Others	O			

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

Table 5.2
PRA Seismic Fragility

Plant Name: Indian Point Unit 2 SSE: Horizontal not identified (g) SSE: Vertical not identified (g)

Hazard parameter: not identified (PGA, Spectral Velocity)

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

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

Cutoff "g": 1.0g

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
Switchyard	0.30	0.25	0.50	n/a	0.09		
Refueling Water Storage Tank	0.61	0.29	0.32	n/a	0.22		
Emergency Diesel Generator Control Panels	0.65	0.28	0.44	n/a	0.20		
Unit 1 Superheater Building Stack	0.73	0.62	0.21	n/a	0.19		
Boric Acid Storage Tank	0.80	0.28	0.30	n/a	0.31		
Component Cooling Water Surge Tank	0.90	0.30	0.37	n/a	0.30		
Reactor Vessel Internals	1.08	0.24	0.30	n/a	0.47		
Containment Fan Coolers	1.11	0.31	0.21	n/a	0.47		
Condensate Storage Tank	1.13	0.29	0.32	n/a	0.41		
Service Water Pumps	1.23	0.26	0.33	n/a	0.46		
Cable Trays	1.23	0.30	0.28	n/a	0.31		

Table 5.3
PWR Accident Sequence Overview Table

Plant Name: <u>Indian Point Unit 2</u> For Seismic PRA Only <u>1 Sheet of 1</u>

#	Sequence	PDS	CDF	Init. Event	Lost Supports	Failed Functions	Attributes
1	OP-IC		6.2×10^{-5}/ry	T-LOOP	Instrumentation & Control	All	RCP seal LOCA
2	OP-CW		2.7×10^{-5}/ry	T-LOOP	CCW	CCW	RCP seal LOCA
3	OP-EP		1.1×10^{-5}/ry	T-LOOP	EAC	All	SBO, RCP seal LOCA
4	OP-SW		9.6×10^{-6}/ry	T-LOOP	EAC, ESW	All	SBO, RCP seal LOCA
5	CW		9.1×10^{-6}/ry	T-CCW	CCW	CCW	RCP seal LOCA
6	OP-RV-IC		3.6×10^{-6}/ry	T-LOOP	Instrumentation & Control	All, scram	ATWS, RCP seal LOCA
7	OP-CT-R		2.1×10^{-6}/ry	T-LOOP		AFW	TIL
8	OP-RV-EP		2.0×10^{-6}/ry	T-LOOP	EAC	scram, All	ATWS, SBO
9	OP-RV		2.0×10^{-6}/ry	T-LOOP		scram	ATWS

<u>Init. Event (Initiator)</u>. 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

<u>Lost Supports</u>: 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)

<u>Failed Functions</u>: 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)

<u>Attributes</u>: At most three of the following: ATWS, BYPASS, TIL, IND-SGTR, SBO, OR HUM (Field may be blank)

Table 5.4
PWR Accident Sequence Overview Table

Plant Name: Indian Point Unit 2 For Fire PRA Only 1 Sheet of 1

#	Sequence	PDS	CDF	Init. Event	Lost Supports	Failed Functions	Attributes
1	Control Room		$7.1 \cdot 10^{-6}$/ry	T-AC	EAC		
2	Cable Spreading Room		$4.3 \cdot 10^{-6}$/ry	T-AC			
3	Switchgear Room		$3.8 \cdot 10^{-6}$/ry	T-AC	EAC	AFW	
4	Electrical Penetration Area		$1.1 \cdot 10^{-6}$/ry	S2		AFW	
5	Primary Water Makeup Area		1.1×10^{-6}/ry	S2	CCW	HPI	

Init. Event (Initiator): One of the following: S1, S2, S3, A, V (-xx), T-LOOP, T-RX, T-TT, T-ATWS, T-UHS, T-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

Lost 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, NH, 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 5.5
PWR Accident Sequence Overview Table

Plant Name: Indian Point Unit 2 For High Winds PRA Only

#	Sequence	PDS	CDF	Init. Event	Lost Supports	Failed Functions	Attributes
1	W02		Tornadoes $1.1 \cdot 10^{-5}$/ry Hurricanes $1.8 \cdot 10^{-7}$/ry Cyclones $7.6 \cdot 10^{-6}$/ry		EAC		
2	W01		Tornadoes $2.5 \cdot 10^{-6}$/ry Hurricanes $3.4 \cdot 10^{-8}$/ry Cyclones $1.1 \cdot 10^{-8}$/ry		EAC	AFW	
3	W18		Tornadoes $1.1 \cdot 10^{-6}$/ry Hurricanes $3.4 \cdot 10^{-7}$/ry Cyclones $3.8 \cdot 10^{-7}$/ry				

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.

Lost 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: Indian Point Unit 2 For Seismic 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	CHPI	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	OP-IC		X					X																																		RCP seal LOCA
2	OP-CW		X					X																																		RCP seal LOCA
3	OP-EP		X					X	X	X	X				X	X	X											X				X	X	X	X							SBO, RCP seal LOCA
4	OP-SW		X					X	X	X	X				X	X	X											X				X	X	X	X							SBO, RCP seal LOCA
5	CW		X					X																																		RCP seal LOCA
6	OP-RV-IC	X	X					X																																		ATWS, RCP seal LOCA
7	OP-CT-R		X	X																																						Transient-induced LOCA
8	OP-RV-EP	X	X					X	X	X	X				X	X	X											X				X	X	X	X							ATWS, SBO
9	OP-RV	X	X	X	X																																					ATWS

Table 5.7
PWR Accident Sequence Detailed Table

Plant Name: Indian Point Unit 2   For Fire PRA Only

		RX		PRIMARY INTEGRITY					PRIMARY INVENTORY INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT										
#	SEQUENCE	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	RI	HUM	NOTES
1	Control Room																									X																
2	Cable Spreading Room																																									
3	Switchgear Room																																									
4	Electrical Penetration Area			X																																						
5	Primary Water Makeup Area							X		X																																

Table 5.8
PWR Accident Sequence Detailed Table

Plant Name: **Indian Point Unit 2** For High Winds PRA Only 1 Sheet of 1

		RX		PRIMARY INTEGRITY					PRIMARY INVENTORY-INJECTION						PRIMARY INVENTORY-RECIRC					SECONDARY INTEGRITY					SECONDARY INVENTORY							CONTAINMENT										
#	SEQUENCE	RPS	BI	PPORV	PSRV	PAD1	PAD2	RCPS	CHP1	HP1	IP1	ACC	AI1	AI2	CHPR	HPR	LPR	AR1	AR2	SGS	SGA	II	MSIV	IB	SG	MFW	NISP	AFW	AM1	AM2	AM3	CS1	CS2	FC1	FC2	IC	CI1	CI2	IGN	RI	HUM	NOTES
1	W02																																									
2	W01																																									
3	W18																																									

6 REFERENCES

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Attachment 2

TECHNICAL EVALUATION REPORT

INTERNAL FLOOD EVALUATION

INDIAN POINT UNIT 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 15, 1998

MEMORANDUM FOR: Mary Drouin, Acting Branch Chief
Probabilistic Risk Analysis Branch
Division of Systems Technology
Office of Nuclear Regulatory Research

FROM: John C. Lane, Senior Reliability & Risk Engineering
Risk Based Regulation & Reliability Section
Probabilistic Risk Analysis Branch
Division of Systems Technology
Office of Nuclear Regulatory Research

SUBJECT: INTERNAL FLOOD INPUT FOR INDIAN POINT 2 IPE REVIEW

The staff has reviewed Chapter 5 of the Indian Point 2 IPE submittal pertaining to internal floods, as requested in the memorandum from Alan Rubin to Mary Drouin, dated June 5, 1998. The licensee calculated that internal flooding increased core damage frequency by 6.7E-06 per reactor year. Based on the review we conclude the internal flood analysis meets the intent of Generic Letter 88-20.

Attached is the staff Technical Evaluation Report which provides additional details and evaluations concerning the flooding analysis. This may be used as input into the Staff Evaluation Report.

cc: M. Cunningham
T. King
A. Rubin

Attachment 2

TECHNICAL EVALUATION REPORT

INTERNAL FLOOD EVALUATION

INDIAN POINT UNIT 2

Attachment

Technical Evaluation Report

Internal Flood Evaluation

of the Indian Point 2 IPE

The licensee performed a flooding analysis which consisted of the following steps:

1. Information gathering
2. Determination of the flood areas
3. Determination of the flood sources and damage mechanisms
4. Structural evaluations and penetration integrity checks
5. Screening analysis
6. Analysis of remaining, unscreened flood scenarios.

After screening, the licensee performed a detailed risk evaluation on nine flood scenarios. Based on these, they calculated a total flood induced CDF of 6.7E-06 per year.

The three most significant scenarios, taken together, represented about 94 percent of the total flood CDF. The remaining flood scenarios, in total, contributed less than 1E-06 per reactor year to core damage.

The largest flood damage state contributor to CDF was a service water system flood attributable to a postulated break in a 3-inch diameter service water pipe located in the emergency switchgear room. Larger break sizes associated with this piping would not be totally accomodated by the drains, and, consequently, damage could result in as early as four minutes after onset of the break, assuming no credit for flood detection or isolation due to the limited time available.

The second highest contributor was based on generic industry flood data for turbine buildings. It consisted of a turbine building flood which resulted in the non-recoverable loss of normal power to the emergency buses due to damage to 6.9 kV buses in that vicinity.

The third highest contributor was based on a fire protection system pipe break in the deluge valve room located in the control building. Flood propagation occurred to the emergency switchgear room via an interconnecting door.

Information Gathering and Determination of Flood Areas--The flood analysis team performed a plant walkdown to inspect all accessible flood areas in the major plant buildings (turbine building, primary auxiliary and auxiliary feedwater building, diesel generator building, control building, service water intake structure, and the fuel storage building.) Flood sources within the containment were not considered because their potential for causing core damage was already considered as part of the LOCA analysis. The walkdown included a review of the plant drawings, such as, piping and instrumentation diagrams, system descriptions, known flooding events and prior flooding analyses performed for previous PRAs. The purpose of the walkdown was to define significant flood sources and to note ingress/egress paths for water, protective features such as sumps, alarms, moats, splash guards, inter-area connections relevant to flood propagation, and vital equipment locations.

Determination of the Flood Sources and Damage Mechanism—The licensee defined two potential flood hazard mechanisms: (1) hazard mechanisms associated with the loss of function of the water-filled equipment, and (2) hazard mechanisms resulting from the flood event itself. Flood scenarios which could not be considered benign or bounded by other floods were identified. The associated area was evaluated for susceptible equipment and the potential for the flood to cause an initiating event was determined. The impact on accident mitigating equipment was also examined.

Structural Evaluations and Penetration Integrity Checks—The licensee considered flood barrier failure modes including: leakage through unsealed doors and hatches, mechanical failures, and failures of doors and seals due to excessive water head. Leakage through wall seams was disregarded since observed leakage rates were low. In general, the total collapse of walls was not considered a significant issue with respect to flooding.

Screening Analysis—In the screening analysis, flood propagation was assumed to occur if a given path existed, unless it required a physical barrier whose failure probability was independent of flood height. In those cases, propagation was assumed to occur with a probability equal to the probability of the barrier failure. Susceptible equipment was assumed to be damaged given that a flood occurred, unless damage could easily be ruled out on the basis of an inadequate maximum flood height and no possibility of spraying. As a result of the screening analysis many potential flood scenarios were screened out because they were not judged to be risk significant, i.e., the core damage frequency was estimated to be much less than 1E-06 per year, or the scenario effects were already included in other parts of the internal events PRA.

Scenarios screened out include maintenance induced floods. To estimate the flood frequency from maintenance actions at full power, plant data was used to determine the frequency of major maintenance on pumps and valves that might require the total disassembly of the pumps or valves. The maintenance action was then coupled with the failure to remove power to the associated isolation valve. This was then followed by a coincident demand on the system (or valve rupture), thereby opening the isolation valve thereby initiating the flood. These floods were screened out for one of the following reasons:

1. The pumps were located in buildings containing no safety-related equipment or equipment required for a plant shutdown, e.g., the fire pumps.

2. The systems being maintained were continuously operated and any leakage would be detected quickly, e.g., the charging and component cooling water pumps.

3. The initiating event frequency of the maintenance induced flood was small compared to a very similar scenario initiated by pipe rupture, e.g., failures of the safety injection valves, RHR valves, or core spray valves, resulting in spilling of the RWST inventory in the primary auxiliary building.

Quantification of Remaining, Unscreened Flood Scenarios—The approach for the flood-induced core damage frequency quantification consisted of the following steps:

1. Determination of the frequency and size of potentially significant flood sources
2. Definition of flood damage states
3. Evaluation of flood growth
4. Quantification of flood induced accident sequence frequencies

Included in the assessment of individual sources of flooding were pipes, tanks, expansion joint failures, and spurious fire system activation. Component leakage and rupture failures were based upon industry data as was the estimate of fire system actuation frequency of magnitude sufficient to disable plant systems or initiate a plant transient.

The consequences of flooding were analyzed by considering the effects of water accumulation, water spraying, environmental conditions, and flood propagation. The parameters considered for the effects of water accumulation included: the flood area volume, the flood source inventory, the flooding rate, the flood propagation rate, and the drain capacity. For water spraying effects, the parameter considered was the proximity of pipes to safety-related equipment and the protection afforded the equipment against spray effects. The effects of environmental conditions were based on a determination of the effects of high pressure and temperature on nearby safety-related components. The effects of flood propagation were based on the propagation into the adjacent flood areas and the susceptibility of the safety-related equipment in those areas.

Each flood damage state was defined in terms of the time at which it would occur after the initial flooding incident together with the important systems which would be damaged. Damage could occur immediately due to spraying or dripping or could later due to rising flood level (flood growth) or propagation to adjacent areas and equipment.

Finally, the flood damage states and core damage frequencies were quantified. This was accomplished by making modifications to the IPE general transient event tree to reflect the specific flood damage states and the associated impacts on related equipment. As indicated previously, the overall CDF resulting from flood was reported by the licensee to be 6.7E-06 per reactor year.

The staff has evaluated the flood analysis provided in the Indian Point 2 IPE submittal. We have determined that the licensee has reasonably assessed the potential effects from internal flooding and therefore conclude that this portion of the IPE meets the intent of Generic Letter 88-20.